Proceedings
of the
St. Michaels Workshop
on

Residual Radioactivity and Recycling Criteria

September 27-28, 1989

St. Michaels, Maryland

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PREFACE

Over the next several decades, many thousands of radioactively contaminated sites in the United States, Japan, and other countries will become candidates for cleanup and decommissioning.

Some sites are already in the process of cleanup -- in the U.S, for example, under DOE and Superfund programs. For other facilities, such as the 500 operational commercial nuclear power plants throughout the world, large-scale decommissioning and cleanup will become increasingly common in the coming decade.

But suitable public health and environmental protection criteria and standards for such decontamination and decommissioning programs are lacking or incomplete in virtually all countries. In their absence, the responsible agencies and industries have had to resort to ad hoc criteria. These often are inadequate and inconsistent, generate confusion, and lack public confidence.

Health risks to some individuals could be significant. And since some radionuclides have long half-lives and high environmental mobility, these risks could be long-lasting and wide-spread. Because decontamination is anticipated to cost many tens of billions of dollars, this is also an issue of substantial economic importance. The lack of established cleanup levels exacerbates this aspect of the problem.

Thus, health protection criteria are needed so that sites can be cleaned up and made available for other uses, either with or without restrictions based upon residual radioactivity.

On September 27 and 28, 1989, the Office of Radiation Programs of the Environmental Protection Agency and the Japan Atomic Energy Research Institute together sponsored a workshop on Residual Radioactivity and Recycling Criteria in St. Michaels, Maryland. Thirty-one government and private sector radiological health experts from the United States participated, and eleven from Japan.

The workshop provided a forum for an exchange of ideas and information among individuals who deal directly with the issue of cleanup standards. Topics for discussion fell into five general categories: Extent of the Clean-up Problem; Impacts of Clean-up Technologies and Economics on Criteria; Health Effects; Desirable Characteristics of Criteria; and Recycling of Materials and Equipment. There was a Panel discussion at the conclusion of the meeting.

Workshop participants made clear their feelings that the presented papers and the debate were highly productive. We are therefore publishing this volume of Proceedings in the hope that it will be of use as a resource document in the further development of criteria for the cleanup of radioactively contaminated sites.

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Introduction

Richard Guimond
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INTRODUCTION

I am particularly pleased to be able to welcome you to this workshop on cleanup criteria for radioactively contaminated sites, sponsored jointly by ORP and the Japan Atomic Energy Research Institute (JAERI). We at ORP have undertaken several projects together with our friends at JAERI, and we have found them to be colleagues of great value. I feel that this workshop is an important part of our growing collaboration.

THE NEED FOR CLEANUP CRITERIA

In the United States, the issue of how to deal with radioactively contaminated facilities is one whose time has clearly come. The press is filled with reports of sites, some privately run and others owned by the government, that are seriously contaminated and will have to be cleaned up to protect the health and well-being of those who live nearby. The spector looms, moreover, of one hundred commercial power plants, many of which will have reached the end of their useful lives early in the 21st century, and will have to be decommissioned. It is estimated that in the U.S., there are perhaps a thousand significantly contaminated sites of various types that must be cleaned up. And the Congress, the Federal agencies, and all aware citizens are concerned.

In the meantime, there has been little cleanup progress to date. One important reason for this is that there are few applicable cleanup standards. There is nothing that says to the owner or manager of a facility where radioactive materials are used, "If you clean your place up to such-and-such a level, you can walk away from it and forget it."

In the absence of suitable public health and environmental protection criteria, cleanup efforts have occurred ad hoc, with widely differing results. Standards that were written for the decontamination of uranium and thorium mill tailing sites (40 CFR 192), for example, are being used as cleanup criteria at Superfund and Department of Energy sites that are contaminated with naturally occurring radioactive materials (where radium is usually the culprit). In some cases, this has required stretching the standards, since they were originally designed for a much more specific situation.

As a consequence of the absence of criteria, cleanup has been postponed, in some cases, in anticipation of standards that might be forthcoming. The cleanups that have been undertaken have frequently been incomplete and inconsistent, have generated confusion, and have lacked public trust.

There is thus a pressing need for clear and unequivocal criteria that will ensure uniform and adequate protection of the public nation-wide. These standards should be consistent with the requirements for the cleanup of other kinds of environmental hazards. And for economic and ethical reasons, most if not all contaminated sites, large and small, will probably have to be made clean enough for release for unrestricted public use. The public deserves nothing less, since we cannot predict how long we can rely on the institutional controls that would be made necessary by less than complete cleanup.

HOW CLEAN IS CLEAN ENOUGH?

At many sites, large amounts of radioactive contamination have become mixed into the soil, sometimes to depths of several meters or more. It may have percolated down into the aquifer, making water unsuitable for home use, for livestock, and for irrigation of crops. It may have become imbedded in the walls and floors of buildings, from which it can not only produce direct external irradiation, but also rub off onto hands and clothing.

After the sites are cleaned up and released for unrestricted use, people will then use the land and groundwater to raise crops and animals, and will raise families and work within the buildings that remain. That raises the critically important central question: How clean is clean enough for such unrestricted public use?

For a carcinogen like ionizing radiation, we have no threshold level below which we can feel totally secure. We don't even have any legislative guidance on what an acceptable level of residual radioactivity would be.

Should standards restrict individual lifetime risk of a fatal, radiation-induced health effect to 10^{-4}, or to 10^{-6}, or to some other level? How important is the ALARA principle in establishing cleanup criteria, and what levels of radiation should be considered Below Regulatory Concern? There are no readily available answers to these fundamental questions.

The problem of cleanup criteria is complicated even further by the special difficulties of disposing of radioactive materials, and by the particular distaste that people harbor for things radioactive.

The disposal problems are unique. Unlike many other hazardous substances, radioactive materials will not decompose or otherwise disappear by themselves. Nor can you simply incinerate the soil and then put it back in place. Radionuclides have half-lives of thousands of years or more, in some cases, and come in a wide variety of chemical forms, often mixed with other hazardous materials. The sheer volume of low level waste projected to come from decommissioning operations is staggering. It has been estimated that there would be one million cubic meters of LLW left over from the decommissioning of our 100 commercial power plants.

That's enough to build a wall one meter thick and three meters high between Washington, D.C. and New York. Perhaps that's what we should do with it.

The sums of money involved are also quite remarkable. Frequently one hears figures like one hundred or several hundred billions of dollars. By contrast, EPA's Superfund budget for the 1989 fiscal year has been $1.6 billion. And, of course, there will be many different organizations and individuals that will be financially affected by the details of any standards and regulations that are set, and ready and able to apply political pressure in support of their needs.

A few areas, such as the Nevada nuclear weapons testing range, are not only highly contaminated with wide-spread radioactivity, but also extremely remote and inaccessible. The costs of a complete cleanup would be very high, and the health benefits marginal. It might be appropriate to perform a partial cleanup, and plan on institutional controls for an indefinite period of time to prevent human intrusion. But is it reasonable to count on these services continuing far into the future?

Another important complicating factor in the establishment of criteria is that the idea of "radiation" tends to evoke emotions in our fellow citizens that are perhaps best described as "intense". People frequently feel that sources of radiation are beyond their control, unnatural, and certainly not freely chosen. Such a situation spells "outrage", and that outrage may well stand in the way of setting good and equitable residual radioactivity standards. We must take into account, and respect, the opinion of the public.

Meanwhile, decommissioning may result in the accumulation of large amounts of slightly contaminated materials and equipment. There are thousands of tons of slightly radioactive copper and nickel and steel which, if properly used, could be of significant economic and societal value. But if released for general circulation, they could cause harm. In addition to the potential health risks, they could be highly damaging to the photographic, microelectronics, nuclear counting, and other sensitive industries. Restricted recycling, on the other hand, such as employing contaminated steel for railroad tracks or in bridges, might fall at the other end of the benefit-risk scale.

A VARIETY OF PERSPECTIVES IS ESSENTIAL

The selection of clean-up criteria for decommissioning and release of radioactively contaminated sites and materials is an extremely complex task, one that requires hard work, the exercise of intelligence, imagination, and good judgment. It is essential that the people involved in the process represent a wide variety of perspectives. We need input from everyone who has anything to do with the problem, from cradle to grave. Let me identify some of the people that I have in mind:

- The people who prepare and employ the materials that cause the problem - They extract the radioactive materials from ores and process it into yellowcake. They build and operate the facilities that produce the radiopharmaceuticals and the industrial radionuclides, the electricity and the nuclear weapons.

- The people who created the problem as it exists now - Some poured their chemical and radioactive wastes into holes in the ground, and others stored it in leaking tanks and forgot about it. Most disposed of it exactly as they were supposed to, in strict accordance with the law. Unfortunately, the law itself was not adequate to deal with the problem.

- The people who now have the problem, and must resolve it - the owners of nuclear power plants that will soon be decommissioned; the Department of Energy, which has awakened to a loud public clamoring about past mistakes at weapons production facilities; and the managers of Superfund, saddled with sites that no longer have any owners or responsible parties.

- The people who will establish cleanup levels for such sites, and figure out how actually to clean them up - They are officials of Federal and State governments and their contractors, and those in organizations in a position to offer expertise and advice, such as the National Academy of Sciences, the National Council on Radiation Protection and Measurement, and the Health Physics Society.

- And finally, the people who must actually carry out the cleanup work, and who will have to ensure, after the fact, that a good enough cleanup job was done - These are the utilities, and the Federal and State agencies, again, with the newspapers and environmental groups looking over their shoulders.

It is important to understand and take into account the perspectives and experiences of all these sorts and groups of people. And for this reason, I am extremely glad that you are here today. I believe that you can play a constructive role in the early stages of developing residual radioactivity criteria. And I believe that this gathering is an excellent collection of the kinds of experts needed to address so pressing a problem.

I thank you for coming, and wish you success in this important workshop.

Session I

Extent of the Cleanup Problem

Department of Energy Environmental Restoration and Waste Management Five-year Plan Environmental Restoration Program

R. P. Whitfield
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ABSTRACT

On September 1, 1989, the Department of Energy (DOE) made available for public comment the first Five-Year Plan for Environmental Restoration and Waste Management. This plan establishes an agenda for compliance and cleanup against which progress will be measured, and it establishes a 30-year goal for the completion of environmental cleanup. Specific implementation plans are being developed by the DOE's field Operations Offices. The Five-Year Plan and Operations Office Implementation plans are "living documents" that will be updated annually. The Environmental Restoration (ER) program, as addressed by the plans, deals with the assessment and cleanup of inactive potential release sites, the decontamination and decommissioning of surplus nuclear facilities, and technology development needed for remediation activities. Preparation of the Five-Year Plan began in March 1989, when a task force was created, and a guidance was issued for field input that provided the basis for the plan. Validated field input was integrated and manipulated electronically to generate the data needed to establish the problem scope, priorities, funding requirements, and other elements of the plan. The data show that the problems within the ER program include approximately 3,700 potential release sites, more than 5,000 vicinity properties connected with the remediation of uranium mill tailings, and approximately 500 contaminated facilities. The estimated funding requirement for all ER activities for the period of 1991 through 1995 is $6.8 billion. In addition, several key needs have been identified while preparing the plan. DOE has developed strategic objectives for ER that include an aggressive applied research and development effort, and it is taking actions to address the problems and needs associated with environmental restoration and waste management. As part of the implementation process, these efforts include participation and review by involved parties.

On September 1, 1989, the Department of Energy (DOE) made available for public comment the first Five-Year Plan for Environmental Restoration and Waste Management. This

plan establishes an agenda for compliance and cleanup against which progress will be measured. This document will be the directive for the development of specific implementation plans by DOE's eight field Operations Offices.

The Environmental Restoration Program (ER) is concerned with the assessment and cleanup of facilities and sites that are no longer a part of active operations. Various amounts and types of wastes have accumulated at these facilities and sites as a result of defense programs, nuclear energy, and energy research program spanning nearly five decades. Included within the scope of ER are Remedial Actions (RA) and Decontamination and Decommissioning (D&D). In addition, technology development and demonstration necessary for the assessment and cleanup of inactive sites and facilities are within the scope of the ER program.

The RA program is concerned with the assessment and cleanup of inactive, potential release sites including burial grounds, spill sites, pits, cribs, lagoons, buried tanks, and uranium mill tailings. Active disposal facilities do not fall within the scope of RA. The tasks associated with ER encompass site discovery, preliminary assessment and inspection, site characterization, analysis of cleanup options, selection of remedy, cleanup and site closure, and site monitoring.

The D&D program addresses the safe caretaking of surplus nuclear facilities until either decontamination for reuse or their complete removal. This includes all tasks connected with assessment and characterization, environmental review, engineering, D&D operations, and closeout.

Preparation of the Five-Year Plan, including the portion dealing with ER began in the early spring of 1989. A task force was created, and a guidance was issued to the eight Operations Offices requesting input on activities proposed during the five-year planning window. The guidance also defined planning areas, established criteria for assigning priority levels, and designed the format and content of input on proposed activities.

Approximately 800 activity data sheets (ADSs) were prepared by the eight Operations Offices for ER activities and were submitted as input to the overall Five-Year Plan. The ADSs were submitted to the task force in the form of a data base diskette as well as hard copy. The input from the field was reviewed by the task force to ensure accuracy, completeness, and conformance with the guidance. The validated data were then compiled into a computer data base by which the information was integrated and managed. Each ADS was assigned a Budget and Reporting (B&R) code and subprogram category (e.g., ER), which allowed the input data to be sorted electronically with respect to program (e.g., Defense Programs), subprogram, category (CERCLA, RCRA, etc.), and phase (assessment or cleanup). The data base also included the assigned priority level, descriptive key words, and funding summaries. Other information provided on the ADS hard copies, but not part of the data base included cost estimates, major milestones, major items of cost, and a statement of the level of confidence in the information presented.

ER activities are organized into four interim priority categories. The priority assignments will be reviewed by DOE on an annual basis and, to the extent that circumstances associated with a specific activity change, its priority may change correspondingly. The priorities are listed below.

Priority 1 includes (1) protecting workers and the public from near-term (i.e., within 5 years) potential health risks, (2) containing near-term off-site spread of groundwater and soil contamination, (3) preventing unnecessary disruption of ongoing assessment and cleanup work, and (4) preclosure surveillance and D&D.

Priority 2 include activities, not otherwise assigned to Priority 1, that are required by in-force agreements or agreements expected to be placed in force during 1991.

Priority 3 includes all activities, not assigned to Priorities 1 and 2, that will best (1) reduce the potential for health and environmental risk, (2) promote regulatory compliance, (3) reduce public concern, and (4) ensure no disruption in DOE's missions.

Priority 4 includes activities not covered under Priorities 1, 2, and 3. Priority 4 is concerned with D&D activities that involve no present imperatives or significant benefits associated with immediate cleanup.

Approximately 3,700 potential release sites have been identified for RA. These site include about 2,480,000 cubic meters of low level waste consisting of discarded materials such as tools, paper, and rags, primarily in burial grounds, to be assessed and remediated as appropriate. In addition, unknown portions of 192,000 cubic meters of pre-1970 buried transuranic (TRU) waste are in inactive sites and as such are within the scope of ER. TRU waste refers to substances contaminated with manmade radioactive elements, principally plutonium, having an atomic number greater than that of uranium, a half-life greater than 20 years, and a concentration greater than 100 nanocuries per gram. Examples of TRU waste include metal, glassware, process equipment, soil, filters, and clothing. Remaining sites consist of hazardous and mixed hazardous and radioactive waste releases. In addition, more than 5,000 vicinity properties are connected with the Uranium Mill Tailings Remedial Action Program. The principal concerns connected with these RAs pertain to groundwater and soil contamination.

Approximately 500 contaminated facilities are included under the D&D effort. Groundwater and soil contamination are associated with only a relatively small number of facilities. The majority of activities involve assessment and cleanup of facilities from which there has been no release of radioactive, hazardous, or mixed substances. The principal concerns pertain to the collection, retention, and ultimate disposal of contaminating substances and debris.

The 30-year goal for ER is to ensure that risks to the environment and to human health and safety posed by inactive and surplus facilities and sites are either eliminated or reduced to prescribed, safe levels. A set of discrete strategic objectives connected with RA and D&D define the overall approach to achieving this goal.

The objectives of RA are to (1) identify inactive, contaminated facilities and sites at DOE nuclear installations, (2) assess these facilities and sites to determine the nature and extent of contamination, (3) confine and contain existing contamination to the extent necessary for minimizing its further spread, (4) provide for negotiated agreements with regulatory schedules for the cleanup of these facilities and sites, (5) ensure that cleanup is carried out in strict compliance with these agreements, and (6) provide long-term monitoring to ensure continuing compliance.

The strategic objectives associated with D&D are to (1) maintain facilities awaiting either decontamination or decommissioning in a manner that limits worker, public, and environmental exposure to potential hazards; (2) assess facilities to determine the nature and extent of contamination; (3) decontaminate facilities designated for reuse to the extent necessary for compliance with approved health and safety standards; and (4) decommission all other facilities in accordance with the requirements set forth in an approved environmental compliance plan.

The role of applied R&D in the strategic approach for ER is to (1) provide an improved technical and economic basis for dealing with environmental and health hazards through development of improved and new assessment and cleanup technologies, (2) reduce the potential for exposure of the public through development of automated remote handling technologies, and (3) broaden the technical base by adapting technologies not previously considered for application to this field. R&D activities that provide a benefit return in a short time frame will be emphasized.

While preparing the plan, several key needs were identified, including (1) a centralized management structure for Environmental Restoration and Waste Management Activities; (2) a cultural transition from a production-oriented mentality to one stressing open communication, clearly understood and demonstrated priorities for environmental stewardship, and accountable management; (3) a national consensus on goals, objectives, and implementation strategy; (4) an aggressive applied research, development, demonstration, testing, and evaluation program; (5) adequate resources including sufficient personnel with proper qualifications to manage and review the work; and (6) consistency in the implementation of environmental regulations.

DOE is taking actions to address the problems and needs associated with Environmental Restoration and Waste Management. Specifically, DOE will (1) comply with environmental and health related laws; (2) develop a national prioritization system with state, tribal, and other public involvement; (3) contain known contamination and vigorously assess the uncertain nature and extent of contamination to enable realistic planning, scheduling, and budgeting; (4) support the establishment of interagency agreements and fulfill the requirements of existing compliance agreements; (5) release health records of DOE employees for scientific analysis; (6) implement waste minimization programs; (7) establish an Applied R&D program involving university research capabilities, industry, national laboratories, and other federal agencies to determine and rank R&D needs and pursue new and improved technologies for minimization and remediation; (8) effect a cultural shift toward clear and open communications; (9) work diligently to achieve congressional support; (10) take innovative steps to develop, motivate, and allocate the human resources needed to implement compliance and cleanup activities; (11) recognize tribal sovereignty and treaty rights related to tribal and ceded lands; and (12) continually examine environmental regulations to ensure that DOE's compliance actions effectively reduce risk to human health and the environment.

Issuance of the first Five-Year Plan initiates an ongoing process within DOE that sets the path for achievement of the goal for completion of environmental cleanup within 30 years. Operations Offices have been directed to develop five-year implementation plans that will include participation and review by involved regional parties in the same manner as the DOE Five-Year Plan. The implementation plans will be used in the management and implementation of actions undertaken by each Operations Office. Future annual updates to the Five-Year Plan, Implementation Plans, and accountability will follow the cycle illustrated in Fig. 1. The second

and subsequent planning/implementation cycles will follow the federal budgeting calendar as shown in Fig. 2.

Fig. 3 illustrates funding requirements estimated for each year of the period from 1989 through 1995 by priority level and phase (assessment or cleanup). The amounts shown for 1989 are those currently appropriated. For 1990, the funds identified are estimated requirements for all activities. The Funding levels shown for 1991 through 1995 are estimates of requirements for funding all RA, D&D, and R&D activities. They do not represent a projection of DOE budgets. The total for this period is $6.8 billion.

Figure 1

THE ENVIRONMENTAL RESTORATION PROGRAM

Included Within The Scope Of Environmental Restoration Are:

- Remedial Actions - Assessment And Cleanup Of Inactive Potential Release Sites
- Decontamination And Decommissioning - Safe Caretaking Of Surplus Nuclear Facilities Until Either Decontamination For Reuse Or Their Complete Removal
- Technology Development And Demonstrations Necessary To Conduct Cleanups

Includes Defense Programs, Nuclear Energy, And Energy Research Facilities

Figure 2

PROCESS USED TO ACCUMULATE AND INPUT DATA
ENVIRONMENTAL RESTORATION SECTION OF THE FIVE-YEAR PLAN

Need Established:	An Agenda For Compliance And Cleanup Against Which Progress Will Be Measured
Guidance To The Field:	Defined Planning Areas, Priority Levels, And Established Format And Content Of Input On Proposed Activities
Field Input:	Approximately 800 Activity Data Sheets (ADSs) Were Prepared By The Field For Proposed Environmental Restoration Activities And Were Submitted As Input To The Five-Year Plan. The ADSs Were Submitted In The Form Of A Data Base Diskette As Well As Hard Copy
Validation Of ADSs:	Input From The Field Was Reviewed To Assure Accuracy, Completeness, And Conformance With Guidance

Figure 3

PROCESS FOR INTEGRATING AND MANAGING DATA
ENVIRONMENTAL RESTORATION SECTION OF THE FIVE-YEAR PLAN

- A Computer Data Base Was Established Utilizing Validated Input From The Field
- The Activity Data Sheet B&R Codes Were The Key To The Accurate And Timely Integration And Management Of Data.
- The Application Of B&R Coding Is Such That Activity Data Can Be Sorted Electronically With Respect To Program, Subprogram (ER), Category (RCRA, CERCLA, Etc), Phase (e.g., Assessment, Cleanup)
- Additional Parameters On The Computer Data Base Include Assigned Priority Level, Descriptive Key Words, And Funding Summaries
- Other ADs Information Utilized In The Preparation Of The Plan But Not Incorporated In The Computer Data Base Include Bases For Cost Estimates, Major Milestones, Major Items Of Cost, And A Statement Of The Level Of Confidence In The Information Presented

Figure 4

CRITERIA FOR PRIORITY RANKING OF
ENVIRONMENTAL RESTORATION ACTIVITIES FIVE-YEAR PLAN

Priority 1:	Required For Near Term Protection Of Workers And General Public From Potential Health Risk, Containing Near-Term Offsite Migration Of Soil/Groundwater Contamination, Ongoing Assessment And Cleanup, Preclosure Surveillance And Maintenance
Priority 2:	Activities Not Assigned P1 Required By Agreements In Place Or Expected In Place By FY 1991
Priority 3:	Activities Not Assigned P1 Or P2 That Will Best Reduce The Potential For Health And Environmental Risk, Promote Compliance, Reduce Public Concern, Ensure No Disruption In DOE's Mission
Priority 4:	Activities Not Included In The Above Consisting Primarily Of Decommissioning And Decontamination Activities Where Immediate Cleanup Is Not Needed

Figure 5

EXTENT AND NATURE OF THE PROBLEM
WITHIN THE SCOPE OF THE ENVIRONMENTAL RESTORATION PROGRAM

With Respect To Remedial Actions, Approximately 3,700 Release Sites

- Approximately 2,480,000 Cubic Meters Of Low Level Waste Consisting Of Discarded Materials Such As Tools, Paper, And Rags, Primarily In Burial Grounds, To Be Assessed And Remediated As Appropriate
- Approximately 192,000 Cubic Meters Of Pre-1970 Buried Transuranic Waste, A Portion Of Which Is Within The Scope Of The ER Program
- Hazardous And Mixed Wastes (Radioactive Wastes Which Contain Hazardous Substances)
- More Than 5,000 Vicinity Properties Connected With The Uranium Mill Tailings Remedial Action Program

The Principal Concerns Connected With The Remedial Actions Pertain To Groundwater And Soil Contamination

There Are Approximately 500 Contaminated Facilities Included Under D&D

- Approximately 400 Defense Programs Installations
- 30 Are Nuclear Energy Facilities
- The Remainder Are Connected With Energy Research

Soil And Ground-water Contamination Are Associated With Only A Small Number Of These And The Majority Do Not Involve The Release Of Radioactive, Hazardous, Or Mixed Substances.

The Principal Concerns Pertain To The Collection, Retention, And Ultimate Disposal Of Contaminating Substances And Debris.

Figure 5 (con't)

EXTENT AND NATURE OF THE PROBLEM
WITHIN THE SCOPE OF THE ENVIRONMENTAL RESTORATION PROGRAM
(Continued)

Several Key Needs Have Been Identified In The Process Of Preparing The Plan Including The Following:

- A Centralized Management Structure For Environmental Restoration And Waste Management Activities
- Cultural Transition From Production Oriented To One Of Open Communication, Clearly Understood And Demonstrated Priorities For Environmental Stewardship, And Accountable Management
- A National Consensus On Goals, Objectives, And Implementation Strategy
- An Aggressive Applied Research, Development, Demonstration, Testing, And Evaluation Program
- Adequate Resources Including Sufficient Personnel With Proper Qualifications To Manage And Review The Work
- Consistency In The Implementation Of Environmental Regulations

Figure 6

HOW THE PROBLEMS AND NEEDS WILL BE DEALT WITH IN TERMS OF THE FIRST FIVE-YEAR PLAN

The Specific Actions Being Taken By DOE To Address The Problems And Needs Associated With Environmental Restoration And Waste Management Are:

- Comply With Environmental And Health Related Laws
- Develop A National Prioritization System With State, Tribal, And Other Public Involvement
- Contain Known Contamination And Vigorously Assess The Uncertain Nature And Extent Of Contamination To Enable Realistic Planning, Scheduling, And Budgeting
- Support The Establishment Of Interagency Agreements And Fulfill The Requirements Of Existing Compliance Agreements
- Release Health Records Of DOE Employees For Scientific Analysis
- Implement Waste Minimization Programs
- Establish An Applied R&D Program Involving University Research Capabilities, Industry, National Laboratories, And Other Federal Agencies To Determine And Rank R&D Needs And Pursue New And Improved Technologies For Minimization And Remediation
- Cultural Change To One Of Clear And Open Communications
- Work Diligently To Achieve Congressional Support
- Take Innovative Steps To Develop, Motivate, And Allocate The Human Resources Needed To Implement Compliance And Cleanup Activities
- Recognize Tribal Sovereignty And Treaty Rights Related To Tribal And Ceded Lands
- Continually Examine Environmental Regulations To Ensure That DOEs Compliance Actions Effectively Reduce Risk To Human Health And The Environment

Figure 7

STRATEGIC OBJECTIVES FOR ENVIRONMENTAL RESTORATION

- 30 Year Goal To Reduce Or Eliminate Risks To The Environment And Human Health And Safety
- Identify Inactive, Contaminated Sites And Facilities
- Assess Nature And Extent Of Contamination
- Confine, Contain Existing Contamination To Extent Needed To Minimize Further Migration
- Provide For Negotiated Agreements For Cleanups
- Ensure Cleanup Conducted In Strict Compliance With Agreements
- Provide Long-Term Monitoring To Ensure Continuing Compliance

Figure 8

ROLE OF RDDT&E
IN THE STRATEGIC APPROACH FOR ENVIRONMENTAL RESTORATION

- Provide Improved Technical And Economic Basis For Dealing With Problems Through Development Of Improved And New Assessment And Cleanup Technologies
- Reduce The Potential For Exposure Of Public
- Broaden The Technical Base By Adapting Existing Technologies Not Previously Utilized In This Fields
- Emphasize Technologies That Provide A Benefit Return In A Short Timeframe

Decommissioning Waste Characteristics

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ABSTRACT

This paper describes the expected general characteristics of wastes produced from decommissioning nuclear facilities. For boiling water reactors and pressurized water reactors, we summarize information extracted from studies performed by Pacific National laboratories under contract to the U.S. Nuclear Regulatory Commission. These nuclear facilities will generate the largest volumes and activities of decommissioning wastes. We also compare these studies with current waste generation information and briefly address decommissioning waste projections for other industries.

INTRODUCTION

Decommissioning wastes will be generated by a broad range of licensees. These licensees include operators of nuclear fuel cycle facilities such as nuclear power reactors, reactor fuel fabrication plants, and uranium hexaflouride conversion plants. Non-fuel cycle licensees include hospitals, medical research institutions, colleges and universities, industrial research laboratories, facilities involved with the production of radiopharmaceuticals, and other industrial users of radioactive material. Wastes produced by these generators will be very diverse in terms of volume, activity, and other physical, radiological, and chemical characteristics. Decommissioning wastes will range from trash that is only suspected of being contaminated to highly radioactive activated structural components from nuclear power reactors.

In this paper, we present some information on decommissioning wastes obtained from studies performed by NRC contractors. We also compare this information with data on low-level wastes (LLW) currently being disposed at LLW disposal facilities.

Our projections of decommissioning wastes are obtained from a series of studies performed by Pacific Northwest Laboratories (PNL) for the Nuclear Regulatory Commission (NRC). Two of these PNL studies summarize and classify projected wastes from decommissioning boiling water reactors (BWR's) and pressurized water reactors (PWR's) [1, 2]. Decommissioning wastes will be generated that will contain significantly more activity than that in wastes normally generated during reactor operation. In contrast, decommissioning wastes

from other fuel-cycle and non-fuel-cycle licensees should be similar to the wastes generated during normal operations by these licensees.

Our data on current waste disposal was extracted from microfiche copies of shipment manifests and computer-generated reports that NRC staff routinely obtain from the two existing disposal facility operators [3 - 24]. The two disposal site operators are Chem-Nuclear Systems, Inc., the operator at the Barnwell, SC disposal site, and U.S. Ecology, the operator of the Beatty, NV and Hanford, WA disposal sites.

The LLW data is presented in terms of NRC's classification system for LLW. NRC's regulations require that waste delivered to a low-level waste disposal site be classified into one of three waste classes -- Class A, B, or C -- depending upon the concentrations of specific radionuclides listed in Tables 1 and 2 of 10 CFR Section 61.55. Concentration limits are lowest for Class A wastes and highest for Class C wastes. The waste classification system is used to place the most restrictive waste disposal conditions on the most hazardous wastes. Wastes exceeding Class C concentrations are considered not generally suitable for near-surface disposal.

Below we present some of the key data relative to decommissioning wastes.

DECOMMISSIONING WASTES FROM NUCLEAR POWER PLANTS

Assuming a reference 1175-megawatt (electric) (MWe) PWR and a reference 1155-MWe BWR, PNL developed estimates of waste volumes and activities generated from immediate dismantlement of the power plants after a 40-year operating life. Each reactor was assumed to have operated at a 75 percent capacity factor, resulting in 30 effective full-power-years of operation.

Table 1 shows the projected waste stream volumes (in m^3) by waste class from dismantlement of the reference PWR. Virtually all the waste volume is Class A. Class C and greater-than-Class C (GTCC) wastes are activated metals from the reactor vessel and internals. The most voluminous waste stream is Class A contaminated equipment and concrete. Evaporator bottoms, resins, cartridge filters, and other dry active waste (DAW) are projected to be both Class A and B wastes.

Table 2 shows the projected activities for the reference PWR. By far, the most active wastes will be the core shroud and the other GTCC core internals. Although representing a small volume, these activated metal wastes dominate the PWR decommissioning activity.

We believe that the PNL projections of the activity in evaporator bottoms, resins, cartridge filters, and DAW are overly conservative. Current waste processing systems generated substantially lower activities then the PNL report predicts.

Tables 3 and 4 present available 1987 disposal data by waste stream. Note that overall concentrations for comparable waste streams are significantly lower than those projected by the PNL report. For example, for PWR resins the PNL report suggests an average concentration of about 740 curies per cubic meter (Ci/m^3). This value exceeds the maximum recommended activity loading (350 Ci/m^3) for organic resins [28] by a factor of two. In actual decommissioning

practice, radioactive waste concentrations are more likely to be similar to process waste concentrations from normal plant operations.

Table 5 presents the projected decommissioning waste volumes for the reference BWR. Note again that virtually all the waste volume is Class A. Class C and GTCC wastes are activated metals from reactor internals. The most voluminous waste stream is the Class A contaminated equipment and concrete. Evaporator bottoms, solidified decontamination solutions, filter sludges, resins, and other DAW are projected to be both Class A and B wastes.

Table 6 shows the projected activities for the reference BWR. By far, the most active wastes will be the core shroud and the other core internals. Although representing a small volume, these activated metal wastes dominate the BWR decommissioning activity. The PNL projections of the activity in evaporator bottoms, filter sludges, resins, and DAW are considered again to be very conservative. Current waste processing systems generate substantially lower activities than the PNL report predicts. Tables 3 and 4 again suggest that overall concentrations of comparable waste streams are significantly less than those projected by the PNL report. In actual decommissioning practice, radioactive waste concentrations are more likely to be similar to process waste concentrations from normal plant operations.

Table 7 shows a summary distribution of the PNL volume and activity data from PWR and BWR decommissioning.

Tables 8 and 9 present data showing the effects of decay on the Class A, B, C and GTCC PWR and BWR decommissioning wastes. These data are based on the radionuclide distributions assumed in the PNL reports. The tables clearly illustrate how the waste contamination is dominated by short-lived radionuclides such as iron-55 (Fe-55: 2.6-year half-life) and cobalt-60 (Co-60: 5.26-year half-life). The activated metals that dominate the Class C and GTCC wastes are also dominated by Fe-55 and Co-60.

DECOMMISSIONING WASTES FROM OTHER TYPES OF FACILITIES

PNL has estimated decommissioning waste volumes for many other fuel cycle and non-fuel cycle facilities [29 - 30]. Table 10 provides a summary of the waste volumes from these facilities. The volumes of decommissioning wastes from non-power reactor facilities will depend on the particular operations of the licensee. The non-fuel cycle waste projections are for equipment and individual components typically used in non-fuel cycle licensee operations.

Decommissioning wastes from non-utility licensees will contain the same radionuclides used during operations, and should also be physically and chemically similar. Because wastes generated during normal operations are relatively low in activity, decommissioning wastes should also be relatively low in activity.

Table 11 presents overall 1987 waste volume and activity data for disposals by industries generating radioactive waste. These data show that, in terms of waste volume and activity, the utilities are the principal generators of radioactive wastes. Except for the industrial sector, other industries generate very little of the activity. Based on our reviews of shipment manifests, the industrial sector activity is dominated by shipments from a few radioisotope production firms. Therefore, with the exception of these firms, the radionuclide concentrations in these wastes are

very small. Radionuclides found in non-utility wastes are quite varied reflecting the many uses of nuclear materials in the commercial sector.

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Table 1
Waste Stream Volumes (m^3) from Dismantlement of a Reference PWR

Waste Streams	Class A	Class B	Class C	GTCC	Total
Neutron Activated					
Core Shroud and Adjacent Metal				133	133
Other Internals	39	60	17		116
Pressure Vessel	222				222
Concrete	707				707
Metal Cavity Liner	14				14
	982	60	17	133	1192
Contaminated Material					
Equipment, Metal, and Concrete Surfaces	16,078				16,078
Process Waste					
Evaporation Bottoms	266				266
Resins		57			57
Cartridge Filters		9			9
Dry Activate Waste	195	88			283
	461	154			615
Total	17,521	214	17	133	17,885

Table 2
Waste Stream Activities (Ci) from Dismantlement of a Reference PWR

Waste Streams	Class A	Class B	Class C	GTCC	Total
Neutron Activated					
Core Shroud and Adjacent Metal				4,784,500	4,784,500
Other Internals	212	5,501	34,300		40,013
Pressure Vessel	19,186				19,186
Concrete	2,121				2,121
Metal Cavity Liner	10				10
	21,529	5,501	34,300	4,784,509	4,845,830
Contaminated Material					
Equipment, Metal, and Concrete Surfaces	997				997
Process Waste					
Evaporation Bottoms	13,812				13,812
Resins		41,998			41,998
Cartridge Filters		5,040			5,040
Dry Activate Waste	234	528			762
	14,046	47,566			61,612
Total	36,572	53,067	34,300	4,784,500	4,908,439

Table 3
Barnwell 1987 Waste Volumes and Activities
by Waste Type and Class

Waste Type	Class A	Class B	Class C	Total
Volume (ft^3)				
Resin	1.920E+5	2.675E+4	4.414E+3	2.231E+5
Solid Combustibles	1.733E+3	1.190E+2		1.852E+3
Solid Noncombustibles	5.158E+4	1.866E+2	2.730E+1	5.179E+4
Filter Media*	2.193E+4	1.160E+3		2.309E+4
Cartridge/Mechanical Filters**	5.140E+3	9.044E+2	1.675E+3	7.719E+3
Solidified Liquids	5.836E+4	7.836E+2		5.914E+4
Equipment, Components	3.849E+2	2.929E+2	9.700E+2	1.648E+3
Biological	5.077E+3			5.077E+3
Incinerator Ash	3.000E+1			3.000E+1
Air Filtration Filters	3.045E+3			3.045E+3
Combustibles and Noncombustibles (Mixed)	5.782E+5	1.016E+3	4.100E+0	5.793E+5
Total	9.175E+5	3.121E+4	7.090E+3	9.558E+5
Activity (Ci)				
Resin	1.663E+4	1.717E+4	1.040E+4	4.420E+4
Solid Combustibles	8.740E+0	1.300E+2		1.387E+2
Solid Noncombustibles	3.009E+2	5.977E+3	7.216E+1	6.350E+3
Filter Media*	3.114E+2	3.489E+2		6.603E+2
Cartridge/Mechanical Filters	1.616E+2	4.456E+2	1.165E+3	1.772E+3
Solidified Liquids***	8.332E+2	2.347E+2		1.068E+3
Equipment, Components	3.510E+1	1.177E+3	1.516E+5	1.528E+5
Biological	1.357E+0			1.357E+0
Incinerator Ash	7.000E-5			7.000E-5
Air Filtration Filters	5.685E+0			5.685E+0
Combustibles and Noncombustibles (Mixed)	1.237E+3	2.742E+3	4.204E-2	3.979E+3
Total	1.953E+4	2.822E+4	1.633E+5	2.110E+5

* Used in liquids and other than resin or cartridges.
** Used in liquids.
*** Includes concentrates and sludges.

Table 4
1987 Volumes and Activities of Richland and Beatty Wastes

Waste Types	Richland, WA Site Vol. (ft^3)	Act. (Ci)	Beatty, NV Site Vol. (ft^3)	Act. (Ci)
Vials	1.500E+1	1.187E-2	1.158E+2	2.352E+0
Dry solid	4.202E+5	1.927E+4	2.826E+5	9.580E+3
Solidified liquid	2.729E+4	2.284E+4	3.578E+4	6.029E+2
Biological waste	2.813E+2	2.606E-2	1.828E+2	5.859E-3
Filter media	3.465E+3	3.455E+2	1.823E+2	6.649E00
Dewatered resin	3.851E+4	4.002E+3	2.354E+3	8.871E-1
Solidified resin	7.184E+3	7.481E+2	7.376E+3	9.026E+2
Absorbed aqueous liquid	2.967E+4	2.412E+2	9.721E+2	1.743E00
Absorbed organic liquid	4.857E+3	1.764E+0		
Aqueous liquid in vials	4.700E+3	9.921E+0	2.250E+1	6.011E-3
Animal carcasses in absorbent	1.203E+4	1.527E+1	1.602E+3	7.931E-1
Gas			2.250E+1	4.051E-3
Compacted dry active waste	5.700E+2	6.264E-1	5.431E+2	2.013E00
Noncompacted dry active waste	2.075E+2	6.478E+0		
Other	7.657E+3	5.299E+0	6.876E+2	1.239E+0
Total	5.566E+5	4.748E+4	3.324E+5	1.110E+4

Table 5
Waste Stream Volumes (m^3) from Dismantlement of a Reference BWR

Waste Streams	Class A	Class B	Class C	GTCC	Total
Neutron Activated					
Core Shroud				47	47
Other Internals	15	15	53		83
Reactor Vessel	8				8
Concrete	90				90
	113	15	53	47	228
Contaminated Material					
Equipment, Metal, and Concrete Surfaces	17,229				17,229
Process Waste					
Evaporation Bottoms	492	148			640
Solidified Decon Solutions	120				120
Filter Sludge, Resins	54				54
Dry Activate Waste	468	210			678
283	1,134	358			1,492
Total	18,476	373	53	47	18,949

Table 6
Waste Stream Activities (Ci) from Dismantlement of a Reference BWR

Waste Streams	Class A	Class B	Class C	GTCC	Total
Neutron Activated					
Core Shroud				6,301,700	6,301,700
Other Internals	750	10,300	239,000		250,050
Reactor Vessel	2,160				2,160
Concrete	180				180
	3,090	10,300	239,000	6,301,70	6,554,090
Contaminated Material					
Equipment, Metal, and Concrete Surfaces	8,490				8,490
Process Waste					
Evaporation Bottoms	1,440	31,200			32,640
Solidified Decon Solutions	105				105
Filter Sludges, Resins	227				227
Dry Activate Waste	562	1,260			1,822
283	2,334	32,460			34,794
Total	13,914	42,760	239,000	6,301,700	6,597,374

Table 7
Summary Distribution of Volume and Activity
Within Reactor Decommissioning Wastes

Class A	PWR	BWR
Volume (m^3) (%)	17,521 (98)	18,476 (97.5)
Activity (Ci) (%)	36,600 (0.7)	13,900 (0.2)
Class B		
Volume (m^3) (%)	214 (1.2)	373 (2.0)
Activity (Ci) (%)	53,100 (1.1)	42,800 (0.6)
Class C		
Volume (m^3) (%)	17 (0.1)	53 (0.3)
Activity (Ci) (%)	34,300 (0.7)	239,000 (3.6)
GTCC		
Volume (m^3) (%)	133 (0.7)	47 (0.3)
Activity (Ci) (%)	4,784,500 (97.5)	6,301,700 (95.5)
Total		
Volume (m^3) (%)	17,885	18,949
Activity (Ci) (%)	4,908,400	6,597,400

Table 8
Decay of Class A, B, and C Wastes From Reactor Decommissioning (Ci)

Time After Disposal (yr)	Class A	Class B	Class C	Total	Total Reduction Factor
Reference PWR					
0	36,600	53,100	34,300	124,000	1
10	9,480	13,500	9,890	32,900	4
50	168	367	1,090	1,630	76
100	67	163	726	956	130
500	3.1	7.3	45	55	2,300
1,000	0.9	1.7	10.2	12.8	9,700
Reference BWR					
0	13,900	42,800	239,000	296,000	1
10	3,830	11,900	65,100	80,800	4
50	184	684	5,310	6,180	48
100	64.7	286	3,500	3,850	77
500	1.2	10.2	231	242	1,200
1,000	0.4	2.8	64.9	68.1	4,300

Table 9
Decay of GTCC Wastes from Reactor Decommissioning

	Reference PWR		Reference BWR	
Time (yr)	Activity (Ci)	Red. Factor	Activity (Ci)	Red. Factor
0	4,780,000	1	6,300,000	1
5	2,530,000	1.9	3,230,000	2
10	1,380,000	3.5	1,710,000	3.7
30	254,000	19	258,000	24
50	159,000	30	147,000	43
100	109,000	44	100,000	63
300	28,300	170	26,200	240
500	8,090	590	7,770	810
1000	1,550	3,100	1,780	3,500

Table 10
Decommissioning Waste Volumes from Non-Power Reactor Facilities

Facility	Waste Volume (m^3)
Research and Test Reactors	160 - 4,930
Fuel Reprocessing	
Non-TRU	3,100
TRU	4,600
Uranium Hexafluoride Conversion	1,259
Uranium Fuel Fabrication	
Equipment, Concrete, etc.	1,100
Calcium Fluoride Waste	29,600
Spent Fuel Storage	
Wet	2,720
Dry Well	6,700
silo	920
Vault	500
Cask	42
Non-Fuel-Cycle	
Fume Hood	4
Glove Box	4
Small Hot Cell	2
Laboratory Workbench (4.6m x 0.9m x 0.75m)	7.2
Sink and Drain	0.4
Room (6m x 10m x 3m)	7.5

Table 11
1987 Low-Level Waste Volume and Activity Sorted by Generic Industry

Variable	Site	Utilities and Electric Services	Hospitals and Health Services	Colleges and Education Services	Government	Industry	Total
Vol. (ft^3)	B*	6.252E+5	9.740E+2	1.091E+4	5.951E+4	2.592E+5	9.558E+5
	R	2.422E+5	2.151E+4	1.827E+4	9.916E+3	2.647E+5	5.566E+5
	Be	7.198E+4		9.900E+2	6.125E+4	1.982E+5	3.324E+5
		9.394E+5	2.248E+4	3.017E+4	1.307E+5	7.221E+5	1.845E+6
		(50.9%)	(1.2%)	(1.6%)	(7.1%)	(39.1%)	
Act. (Ci)	B	2.009E+5	1.149E-1	1.591E+1	6.086E+3	3.975E+3	2.110E+5
	R	1.721E+4	2.420E+1	5.201E+1	9.855E+2	2.922E+4	4.748E+4
	Be	1.632E+3		5.472E-1	5.293E+0	9.463E+3	1.110E+4
		2.197E+5	2.431E+1	6.847E+1	7.077E+3	4.266E+4	2.696E+5
		(81.5%)	(0.009%)	(0.025%)	(2.6%)	(15.8%)	

* B: Barnwell; R: Richland; Be: Beatty.

Site Inventory of Residual Radioactivity in Japan

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ABSTRACT

The types and numbers of facilities in Japan that may require residual radioactivity criteria for decommissioning have been investigated. The quantities of decommissioning wastes were estimated. The characteristics of the residual radioactivity at the facilities are discussed, based on our decommissioning experience.

INTRODUCTION

Applications of atomic energy have been developed in Japan under "the principle of peaceful use" prescribed in the Atomic Energy Basic Law, which came into force in 1956 [1]. Nuclear power generation has grown to be one of the major sources of electricity in Japan. Radioisotopes and particle accelerators have come into wide use. Most nuclear fuel facilities are coming into a practical and commercial stage from an experimental stage.

No practical scale facility has undergone decommissioning. However, some research reactors and facilities using radioisotopes have been dismantled. Decommissioning of large scale facilities is expected to begin in the near future.

In the present paper, an inventory of contaminated sites, characteristics of the residual radioactivity, and some experiences with decommissioning are reported.

FACILITIES AND THE CHARACTERISTICS OF RESIDUAL RADIOACTIVITY

Reactor

There are four kinds of reactors: commercial power plants, research reactors, critical experimental facilities and developmental reactors. The number of power plants has been increasing since 1966. Thirty-six power plants (17 BWRs, 18 PWRs and one GCR) generate 27,000 MW, and 12 power plants are under construction, as shown in Figure 1 [2]. The electricity generated by nuclear power plants was 25% of the total electricity demand in Japan in 1986.

Assuming that a power plant operates for 40 years and is mothballed for 10 years after shutdown, it is expected that decommissioning undertaken seriously will begin in 2020, and that up to five power plants will be decommissioned every five years. This decommission will yield a large amount of radioactive waste.

Ohta reported the estimated amount of radioactive waste produced in decommissioning a PWR and a BWR (1100 MW), assumed to be closed for 10 years after shutdown, as shown in Table 1 [3]. Radioactive concentration in the most radioactive waste is lower than 10^{-4} Ci/ton. Radioactive metal waste and concrete waste are estimated to be 30,000 tons (5.9% of total radioactive wastes) and 500,000 tons (93%), respectively for BWR, and 30,000 tons (6.1%) and 460,000 tons (91%), respectively for PWR.

Based on the data in NUREG/CR-0672 [4], the Japan Power Demonstration Reactor (JPDR) Research Committee also estimated in detail the quantities of materials contained in a reference BWR plant, as shown in Table 2 [5]. The table indicates that about 1,500 tons of nonferrous metals will be generated from the decommissioning of a BWR power plant. The differences of the estimated quantities are assumed to be caused from the differences in calculation conditions such as history of operation, decontamination factors and extent of evaluation of quantities.

The data obtained from the JPDR decommissioning project are useful for estimating the amount and characteristics of the radioactive waste from the decommissioning of a commercial power reactor. JPDR is a light-water-cooled boiling water reactor which operated at a power of 45 MW for 10 years and at 90 MW for 1 year, and was shutdown in 1976, as shown in Table 3. The reactor is under decommissioning at the present. Table 4 and Figure 2 show the estimated amount of radioactive wastes from the JPDR decommissioning. In these statistics, amounts of uncontaminated materials are not included. Metal waste is estimated to be 1,840 tons (1,640 tons of contaminated metals and 200 tons of activated metals). Approximately 85% of the waste have radioactivity concentrations lower than 10^{-5}Ci/ton. The amount of the metal waste from JPDR was estimated to be one sixteenth of that from a typical 1,000 MW power plant, which is about twenty times the size of JPDR. Major metals involved in JPDR are carbon steel and stainless steel. Also, a small amount of copper waste comes from cables and heat exchangers. There are two kinds of residual radioactivity, contamination and activation. Contaminated concrete was easily cleaned up. By contrast, activation products are widely distributed in the concrete.

There are 11 research reactors, most of which were built in the 1950's. A research reactor, JRR-3 was dismantled and is being converted into a new type of research reactor. A large amount of concrete contaminated with tritium was generated from the dismantling of the reactor. Since tritium is volatile and easily permeates concrete, it spread inside the whole facility and diffused into the concrete. It was difficult to determine the residual activity level below which a cleanup is not required.

There are 4 developmental reactors, JPDR, JOYO (a fast breeder experimental reactor, 100 MW in 1979), FUGEN (an advanced thermal reactor, 557 MW in 1977) and MONJU (a prototype fast breeder reactor, 714 MW which is under construction).

There are seven critical experimental facilities. In these facilities, the residual activity is generally limited to several components and therefore the amount of radioactive waste from dismantling is relatively small.

Nuclear fuel cycle facilities

A demand to develop nuclear fuel cycle facilities is increasing in Japan. The amount of uranium deposits in Japan is an estimated 4,000 U_3O_8 tons. All uranium used in reactor fuel is imported from foreign countries. At two sites, Ningyo-type and Tohko, uranium was mined and milled experimentally. However, these sites are closed at present. A uranium mill facility at Tokai had been operating from 1959 to 1970, and the facility was dismantled in 1978. A pilot plant for milling and conversion started operation in 1982, producing UF_6 at a rate of 300 tons/year. A pilot uranium enrichment facility using the centrifuge method started operation at a rate of 50 ton SWU/year in 1979. On the basis of this operation experience, a prototype plant will come into operation in 1989. A commercial uranium enrichment plant is scheduled to be built in 1995. A large amount of metal materials contaminated with alpha emitters is expected to be produced from dismantling of the uranium enrichment plant.

There are two uranium conversion Plants. One facility started operation in 1962 and processed at a rate of 475 tons of Uranium per year (tU/y). The other started in 1980 and processes 550 tU/y. There are four uranium processing facilities with total processing capacity of 2,000 tU/y.

A pilot plant for fuel reprocessing started treatment at a rate of 0.7 tU/d in 1970. Construction plans for a commercial fuel reprocessing plant are in motion. There is no plan for decommissioning a commercial scale fuel cycle facility in the near future.

However, some experimental facilities have been closed and dismantled. A distinctive feature of nuclear fuel cycle facilities is that they may be contaminated with alpha emitters. Soil produced from uranium mine contains naturally occurring radioactive materials such as U, Ra, Th. The soil was disposed of according to the Mine Safety Law. In a fuel processing facility, floors and walls, especially ventilation systems, were slightly contaminated with U. The dismantling experience indicated that data of incidental contamination, as well as specifications of the facility, were very useful for dismantling plans.

Facilities using radioisotopes

The trend in the number of facilities which have licenses to use radioisotopes or particle accelerators is shown in Figure 3. The number of licenses has increased from 200 in 1959 to 4,257 in 1988. More than 80% of these facilities (4,257 facilities) are licensed to use only sealed radionuclides. For the facilities in which unsealed radionuclides are handled, there is a possibility of contamination with radionuclides. Table 5 shows the amount of radionuclides used in the facilities. The number of facilities using unsealed radionuclides is 82 for medicine, 286 for education and 391 for research. Major radionuclides are Ga-67, Tc-99m and Pm-149 for medicine, H-3 and P-32 in education, and H-3 and C-14 for research. Important nuclides which should be considered in decommissioning are relatively long-lived nuclides, with half-lives longer than 1 year, such as H-3, C-14, Se-75, Pm-147.

JAERI has experience in decommissioning a large radio-chemical laboratory, to convert it into a laboratory where no radionuclides are to be used. In the radio-chemical laboratory, a distinguishing characteristic of residual radioactivity is contamination with H-3 and C-14 through air contamination.

Particle accelerators

The number of particle accelerators in use as of March 31, 1988 is shown in Table 6, classified from the point of view of accelerator types and organizations [9]. The total number of accelerators is 748. The number of Linear Accelerators or LINACS is 497, which is two thirds of the total. The numbers of Cockcroft-Walton, Van de Graaff and cyclotron accelerators are 85, 46 and 34 respectively. Furthermore, Synchrotron Orbital Radiation (SOR) facilities (three in operation and three others are under construction) and experimental fusion facilities may also be categorized as particle accelerators.

The electron accelerators with energy lower than 1 MeV which are commonly used in industrial processing do not come under "radiation generators" in the law concerning prevention of radiation hazard due to radioisotopes, etc. [10]. Thus, these accelerators may be sold to scrap markets free from regulatory restrictions.

It is expected that electron accelerators with energy lower than 10 MeV and positive-ion accelerators with energy lower than 1 MeV also can be reused or recycled, because induced activity produced by these accelerators is absent or negligible [11, 12, 13]. According to the inquiry carried out by the High Energy Physics Research Institute (KEK) [14], about half of all electron accelerators have energies greater than 10 MeV, and approximately 80% of the positive-ion machines have energy greater than 1 MeV. Fusion experimental facilities also produce neutrons and thus induce activity.

For these relatively high-energy accelerators, radioactivity is diluted and dispersed. The radioactivity concentration is relatively low in most parts of the machine except in components such as targets, target holders and positioning mechanisms, slits and collimators [11]. Thus most components are expected to be potential candidates for reuse and recycle. It should be reasonable to reuse or recycle a magnet (carbon steel), coil (copper or titanium), cyclotron vacuum chamber (stainless steel, aluminum alloy or aluminum) and linac drift chamber (stainless steel or aluminum).

For synchrocyclotrons of greatest energy (approx. 730 MeV), the total mass is 8,900 tons [9]. Table 7 lists dominant radionuclides shown in the materials of particle accelerators [11, 14, 15, 16, 17]. By comparison with fuel cycle facilities, accelerators do not produce alpha-emitters, but rather small amounts of short-lived radionuclides. The dominant radionuclide in aluminum is Na-22. In concrete, Na-22, Co-60, Eu-152 and Eu-154 are important in the long term. In ferrous materials, Mn-54, Fe-55 and Co-60 are dominant nuclides. In copper, Co-60 and Zn-65 are important.

DISCUSSION

An inventory of residual radioactivity sites in Japan and their characteristics were reviewed in previous sections. There is no experience in decommissioning a commercial facility which is supposed to generate a large amount of radioactive wastes. But decommissionings of small scale experimental facilities are reported.

According to these experiences, the most practical way to reduce residual activity is to prevent components and building structures from being contaminated, and to remove

contamination when present. It is then important to give consideration to contamination controls and simplification of decommissioning in designing the facility.

It is also essential to keep detailed documents concerning contamination incidents as well as maintenance and/or remodeling records.

A large amount of residual radioactivity in a nuclear reactor is generated both by contamination and by activation. Uranium mine and milling facilities produce soils containing natural radioactive materials. Fuel processing and fabrication plants yield enriched uranium contamination. The decommissioning of fuel concentration facilities is expected to generate a large amount of metal materials contaminated with alpha emitters. A reprocessing plant is contaminated with alpha emitters, fission products and activation products. Major residual radioactivity in a facility using radioisotopes are to be distinguished from contamination with long lived nuclides such as H-3, C-14. Some particle accelerator facilities may yield activation products.

Facilities which generate a large amount of useful metal materials from decommissioning are nuclear reactors, fuel concentration plants, fuel reprocessing plants, and particle accelerators.

The residual radioactivity can be classified into three groups: natural radioactivity in the soil, contamination and activation. The methods of cleanup will be selected depending on the characteristics of the residual radioactivity. When natural radioactivity in soil is homogeneously distributed, a cleanup is called recovery rather than decontamination. The components contaminated with nonvolatile materials are easily cleaned up because such contamination is limited to the surfaces of a facility or of equipment. However, the components contaminated with volatile materials such as H-3 are not easily cleaned up, because the contamination spatially distributes inside the facility and diffuses into materials. Components contaminated with activation products are also difficult to reduce in activity. Proper clean-up methods should be selected depending on the material and the characteristics of the residual radioactivity.

Safety assessment for reuse of land and/or a facility should be conducted depending on the characteristics of the residual radioactivity. It is also essential to choose a proper plan for recycling materials, depending on the characteristics of the residual activity.

It is therefore necessary to give consideration both to the method of recycling or reuse and to the characteristics of the residual radioactivity in establishing residual radioactivity criteria.

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Table 1 Estimated quantities of wastes from decommissioning of BWR and PWR

Radioactive Cocentration (Ci/t)	BWR		PWR	
	Metal	Concrete	Metal	Concrete
$> 10^{-3}$	2,700	540	1500	1,000
10^{-4}–10^{-3}	3,240	540	9,500	0
$< 10^{-4}$	30,780	502,200	30,500	457,500
Total(ton)	36,720	503,280	41,500	458,500

Source: K.Ohta,(1985)

TABLE 2 ESTIMATED QUANTITIES OF MATERIALS CONTAINED IN A REFERENCE BWR PLANT (1155MWe)

Activity concentration level \ Materials			Equipments and Pipings						Structures and Buildings					TOTAL (Equipments and Pipings + Structures and Buildings)	
			Carbon Steel	Low-Alloy Steel	Stainless steel [1]	Non-ferrous metal [2]	TOTAL		Concrete	Rebar [3] (Carbon Steel)	Structural Steel	TOTAL			
Radio-active	≧10^3 Ci/t	Activated	0	0	67	0	67	67	0	0	0	0	0	67	67
		Contaminated	0	0	0	0	0		0	0	0	0		0	
	10^3 ~10^2 Ci/t	Activated	0	0	5	0	5	11	0	0	0	0	0	5	11
		Contaminated	0	0	6	0	6		0	0	0	0		6	
	10^2 ~10^1 Ci/t	Activated	0	148	32	0	180	279	0	0	0	0	0	180	279
		Contaminated	69	0	30	0	99		0	0	0	0		99	
	10^1 ~ 10 Ci/t	Activated	0	0	0	0	0	1,286	0	0	66[4]	66	66	66	1,352
		Contaminated	209	568	509	0	1,286		0	0	0	0		1,286	
	10 ~10^{-1} Ci/t	Activated	0	0	0	0	0	3,687	0	0	0	0	1,079	0	4,766
		Contaminated	1,724	29	519	1,415	3,687		891	0	188[4]	1,079		4,766	
	10^{-1}~10^{-2} Ci/t	Activated	0	0	0	0	0	931	199	0	7[4]	206	1,445	206	2,376
		Contaminated	875	0	0	56	931		1,239	0	0	1,239		2,170	
	10^{-2}~10^{-3} Ci/t	Activated	0	0	0	0	0	0	0	0	0	0	0	0	0
		Contaminated	0	0	0	0	0		0	0	0	0		0	
	10^{-3}~10^{-4} Ci/t	Activated	0	0	0	0	0	1,663	0	0	0	0	0	0	1,663
		Contaminated	333	1,330	0	0	1,663		0	0	0	0		1,663	
	Activity-level Unidentified	Activated	0	0	0	0	0	2,953	0	0	0	0	19	0	2,972
		Contaminated	1,282	220	1,431[5]	20	2,953		0	0	19	19		2,972	
	Sub-total	Activated	0	148	104	0	252	10,877[6]	199	0	73	272	2,609	524	13,486
		Contaminated	4,492	2,147	2,495	1,491	10,625		2,130	0	207	2,337		12,962	
Non - radioactive			Not available in NUREG/CR-0672 [8]						378,572[7]	20,567	2,905	402,074		402,074	
TOTAL			4,492	2,295	2,599	1,491	10,877		380,901	20,567	3,215	404,683		415,560	

(unit : ton)

SOURCE : Report on the Recycle and Reuse of Components from Decommissioning of Nuclear Facilities 1988., JPDR Research Commitee on Recycling and Reuse of Components from Decommissioning of Nuclear Facility

— This table is based on NUREG/CR-0672 and is supplemented with data of domestic power plants.

1) For recent BWR plants in Japan, Inconel is used for some equipments (e.g. decontamination-effluent concentrator : approx. 13 tons) instead of stainless steel.
2) That is aluminium (for pressure-vessel-lid insulation material and pipings), titanium and copper alloy (for condensor pipings).
3) Quantities of reinforce-bar of cooling tower is not available in NUREG/CR-0672. It is assumed to be 100kg/m^3 (1,858 tons) using data of other buildings.
4) Made of low-alloy steel.
5) Includes charcoal filters which consists of stainless steel and charcoal (weigh approx. 33 tons).
6) Quantities of some equipments are not summed up to the total.
7) Includes cooling tower which consists of concrete and asbestos (weigh approx. 40,000 tons).
8) According to the data estimated by power generation company personnels, total mass is estimated to be 30,780 tons.

Table 3 Major specifications and operation history of JPDR.

Major Specifications		
Reactor Type	BWR	
Power	JPDR-I, 45 MWt JPDR-II, 90 MWt	
Reactor Pressure Vessel		
	Material	carbon steel internally clad with SUS
	Height 8m, Diameter 2m, Wall Thickness 7cm	
Biological Shielding		
	Materical	reinforced concrete with carbon steel liner
	Thickness	1.5 - 3m
	Ion bars	max. dia. 29mm iron bar ratio 120-170 kg/m^3
Containment Vessel		
	Height 38m, Diameter 15m	
Operation History		
	Total Operation Time	17,000 hours
	Total Output	21,500 MWD (1.3FPY)

(Source : Ishikawa et al, 1987)

Table 4 Kinds and activity of radioactive waste generated from the JPDR dicommissioning.

Kinds of Radioactive Waste		Activity (Ci)	Weight (Ton)
Activated Components	Core Internals Control rods, Core shroud, etc.	4,500	20
	Pressure Vessel	46	110
	Biological Shield Concrete	12	1,350
Contaminated Components	Components	4.7	1,640
	Concrete	0.2	830
	Resin, etc.	0.5	130
Total		4,600	4,100

(Source : Hoshi et al, 1987)

Table 5 Quantities of unsealded radioisotopes used in fiscal year 1987. (Source: Statistics on the use of radiation in Japan, 1988)

Major Nuclides (unit) \ Category of Organizations	Total	Hospitals & Clinics	Educational Organizations	Research Institutions	Industrial Firms & Other Organizations
^{3}H (mCi)	78 190	1 310	17 480	42 920	16 480
^{14}C (mCi)	20 850	3 380	510	14 760	2 200
^{32}P (mCi)	19 890	540	7 550	11 460	340
^{35}S (mCi)	4 550	190	2 100	2 080	180
^{45}Ca (mCi)	660	20	350	250	40
^{51}Cr (mCi)	6 480	1 220	3 120	1 980	160
^{59}Fe (mCi)	290	170	10	80	30
^{67}Ga (mCi)	422 470	422 450	(2)	(6)	20
^{75}Se (mCi)	620	610	(5)	(5)	(1)
^{81m}Kr Generator (mCi)	21 650	21 650	—	—	—
^{85}Kr (mCi)	5 480	—	—	60	5 420
^{99m}Tc Generator (mCi)	4 047 860	4 044 570	350	720	2 220
^{99m}Tc Solution (mCi)	1 271 100	1 270 240	90	220	550
^{111}In (mCi)	10 410	10 320	50	10	30
^{123}I (mCi)	159 600	159 580	(3)	(6)	20
^{125}I (mCi)	17 830	8 560	3 390	4 350	1 530
^{131}I (mCi)	102 840	101 490	460	690	200
^{133}Xe (mCi)	1 015 120	1 014 540	—	210	370
^{147}Pm (mCi)	5 120 000	—	—	—	5 120 000
^{201}Tl (mCi)	308 470	308 410	(8)	(2)	50

Table 6 Number of radiation generators in use (as of March 31, 1988)

Radiation Generators \ Category of Organizations	TOTAL (Ratio %)	Hospitals and Clinics	Educational Organizations	Research Institutions	Industrial Firms	Other Organizations
TOTAL	748	441	46	126	134	1
(Ratio %)	(100)	(59.0)	(6.1)	(16.8)	(17.9)	(0.1)
Cyclotrons	34 (4.5)	8	—	11	15	—
Synchrocyclitrons	1 (0.1)	—	—	1	—	—
Synchrotrons	11 (1.5)	—	—	7	4	—
Linear Accelerators	497 (66.4)	398	5	24	70	—
Betatrons	43 (5.9)	32	2	2	6	1
Van de Graaff Accelerators	46 (6.3)	—	13	27	6	—
Cockcroft-Walton Accelerators	85 (11.6)	—	22	39	24	—
Transformer-type Accelerators	22 (3.0)	—	3	14	5	—
Microtrons	9 (1.2)	3	1	1	4	—

SOURCE MATERIAL : STATISTICS ON THE USE OF RADIATION IN JAPAN 1988 Edited by Nuclear Safety Bureau in Science and Technology Agency

Table 7 Major radionuclides commonly identified in materials irradiated around accelerators.

Irradiated Material	Radionuclides	
	High-energy electron accelerators(≧35MeV)	Positive-ion accelerators
Aluminium	^{22}Na	^{22}Na
Concrete	^{22}Na	^{22}Na , ^{45}Ca , ^{54}Mn , ^{57}Co , ^{60}Co , ^{152}Eu and ^{154}Eu
Iron , Steel	^{22}Na , ^{54}Mn and ^{55}Fe	^{54}Mn , ^{55}Fe and ^{60}Co
Copper	^{22}Na , ^{54}Mn , ^{55}Fe , ^{60}Co and ^{63}Ni	^{54}Mn , ^{57}Co , ^{60}Co and ^{63}Ni

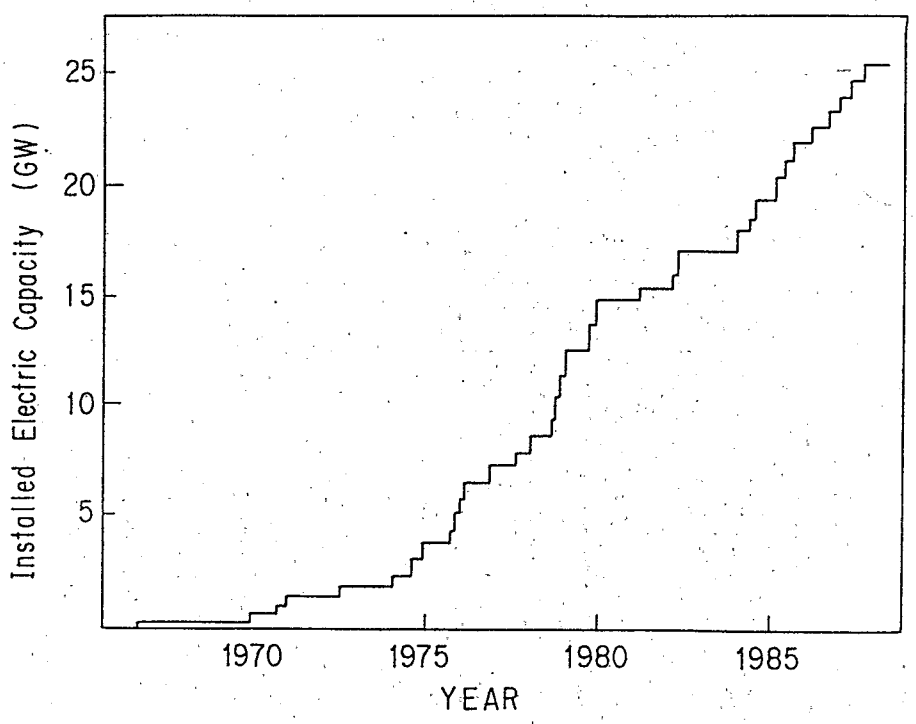

Figure 1 Trend of the installed electric energy capacity on nuclear power plantsin Japan.(Source: Genshiryoku Pocketbook, 1989)

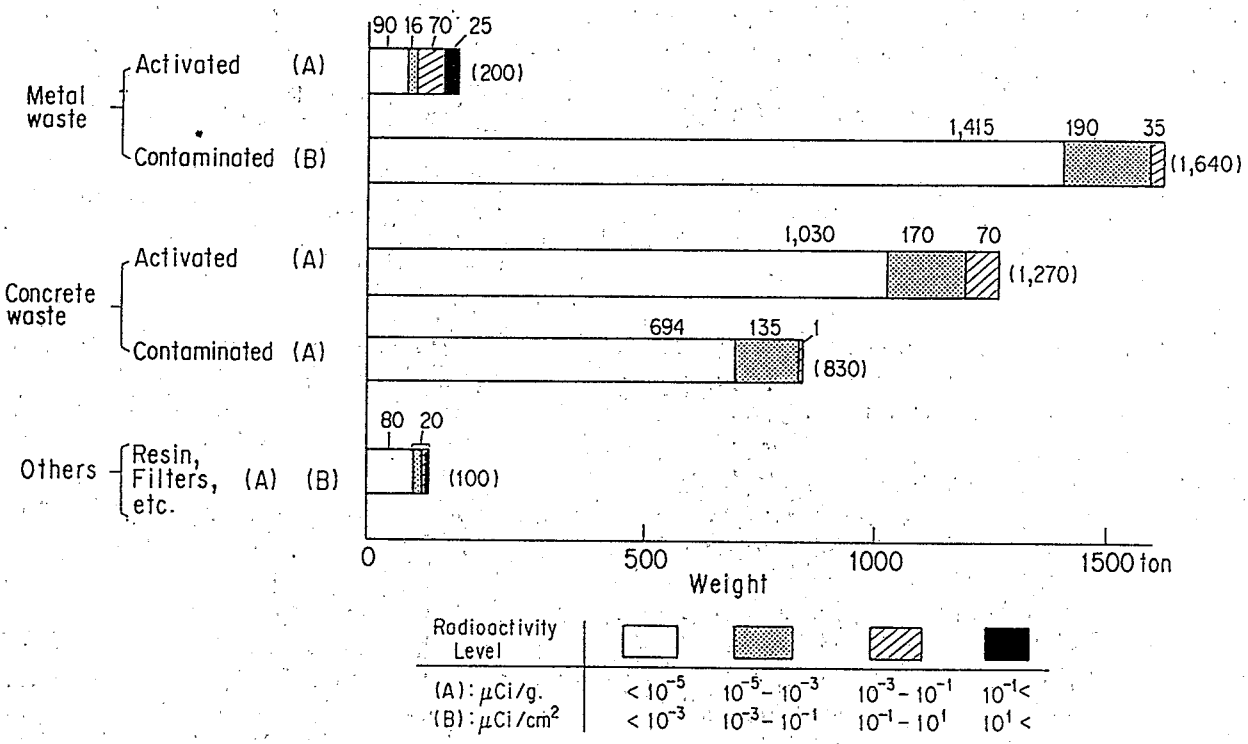

Figure 2 Estimated amount of radioactive solid waste generated from the JPDR. (Source: Hoshi et al, 1987)

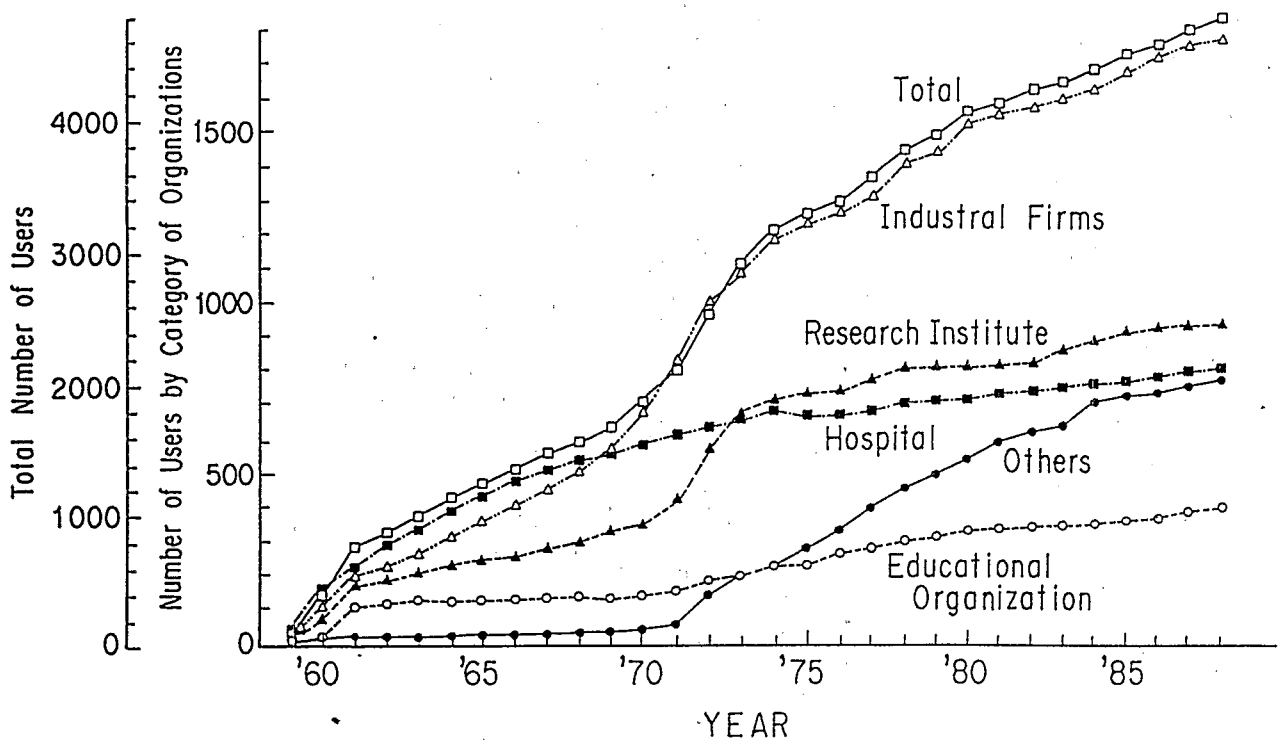

Figure 3 Changes with the year in number of users of radioisotopes in Japan. (Source: Statistics on the use of radiation in Japan, 1988)

Session II

Impacts of Cleanup Technologies and Economics on Criteria

Limitations of Cleanup Technologies

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ABSTRACT

This discussion will include the limitations of cleanup technologies used in the typical processes of decontamination and decommissioning of nuclear facilities. The key issue is to determine the primary objective of decontamination campaigns, namely, to achieve free release of equipment or structures, or to reduce exposure to workers. A thorough radionuclide characterization and contamination survey is a mandatory prerequisite to the development of a properly planned program. The topics to be addressed are the problems associated with performing effective decontamination without increasing LLW volume, radiation exposure or cost; redeposition of contaminants, prevention of cross contamination, process limitations (corrosion, temperature, pressure, application time, pH, flow rate, and waste volume), and actual effectiveness versus vendor's reported effectiveness. Many decontamination programs have failed to meet their objectives because of inadequate preplanning to characterize the contamination and identify the process limitations.

INTRODUCTION

The beginnings of the nuclear era concentrated on processes. From research and development in nuclear technology and weapons production facilities, the decontamination of these facilities was considered a relatively minor concern. However, as the cost of radioactive waste disposal increased rapidly and waste disposal space became scarce, the interest in decontamination technologies grew and decontamination processes were developed to meet this new demand.

Contamination sources included residual radioactivity in the mining and milling processing of uranium and thorium, radium usage in industry for luminescent instruments, weapons production plants, medical facilities, research laboratories and other processes. The development of nuclear power facilities created perhaps one of the largest area for growth of decontamination technology as the utility industry tried to clean up older plants to minimize the areas of each facility that are considered radioactively controlled areas. The cleanup processes spawned new industries and

products aimed at quickly and cost effectively reducing the levels of contamination to either unrestricted use levels, or to levels permitting operating and maintenance without extensive personnel radiological and respiratory protection. Decontamination process and equipment vendors offered their services promising rapid application techniques and high decontamination factors. In some cases these processes were only developed in the laboratory and never field tested prior to their application at a contaminated facility. Not surprisingly, the results were disappointing and in many cases the utility/vendor relationship ended in total dissatisfaction. The primary failure in most of these cases was the lack of a planned decontamination program based on an extensive characterization and identification of the program objectives. This paper will address the limitations of decontamination technologies relative to the removal of residual radioactivity and the possible recycling of materials.

FACILITY CHARACTERIZATION AND PROGRAM OBJECTIVES

The development of a decontamination program must begin with a thorough radiological characterization of the facility to identify the radionuclides present, the concentrations in the various systems or structures, the nature of the contamination as to its inherent adherence to the contaminated surface and the chemical composition of any corrosion product binding the contamination within the corrosion layer. The characterization process is often overlooked, or performed superficially without a clear understanding of the importance to the development of the decontamination program objectives. It is important to realize that in many cases the failure of a decontamination program is due to a chemical problem, not a radiological problem. The management pressure and excitement of getting started and seeing progress is often weighed as more important than carefully developing a realistic program objective that available or developed decontamination processes can satisfactorily achieve.

The characterization pro usually requires several man-months of effort to plan and implement in the field. For internal system characterization, the system piping or components should be accessed (opened) to obtain scraping samples and smears to completely identify the radionuclides and concentrations with in the system. At the same location, external radiation measurement should be made to correlate external exposure readings to internal contamination. By this technique the extent of contamination can be readily estimated using external survey techniques to bound the scope of the decontamination project. For external contamination on structures, concrete cores or steel scrapings should be taken and analyzed for radionuclide content and concentration. This data can be correlated to external survey measurements to determine the extent of contamination on structure surfaces.

Once this characterization data has been analyzed radiologically, chemically and mechanically (for adherence to the surface), a realistic program objective can be developed. The management decision can then be made whether it is feasible to achieve complete decontamination to release the material for unrestricted use and disposal in local landfills or recycling, or whether to reduce radioactivity levels to minimize exposure to workers performing operation or maintenance activities.

There have been far too many decontamination and decommissioning projects where decontamination projects were started based on random and inadequate characterization data. The project had to be stopped, re-characterized, planned and re-started at additional cost never factored in to the original budget.

DECONTAMINATION PROCESSES

In response to the need for effective decontamination processes, several techniques have been developed for specific decontamination tasks. A brief discussion of these techniques is included here as examples of typical methods currently being used. These same examples will serve as focal points to discuss limitations in these and other processes from lessons learned in their application.

High Pressure Water Lancing

This technique directs water through a lance with a jet or fan nozzle at pressures of about 2,000 to as high as 40,000 psi. The erosion action removes surface contamination from steel and concrete surfaces. The process uses 3 to 5 gpm at about 2,000 psi, and about 5 to 10 gpm at 40,000 psi. The operator reaction forces above 10,000 psi are about 60 pounds or more, and result in operator fatigue for prolonged use. At higher pressures the lance cannot be hand-held, and remote/manual or automatic machine must be used. This technique is often used to reduce overall contamination levels on contaminated surfaces rather than to achieve free release of the material.

While this process is very effective in local surface decontamination, there are several drawbacks that limit its usefulness. Operator fatigue was already mentioned. The copious quantities of water used result in a significant radioactive waste management problem. The rapid generation of wastes requires a large waste processing system and associated operators to dispose of the wastes. These costs are often not considered in evaluating the effectiveness of this technique. Portable water filtration units are available to process the water and removed contamination, and to recycle the water through the lance. For lance pressures up to a few thousand pounds pressure, the filtration unit consist of cartridge filters in a pressure housing. The filtration capacity is sufficient to remove steel and concrete particulate, and to recycle the water through the pump system successfully. However, at higher pressures and particularly at pressures as high as 40,000 psi, ultra-filters are needed to remove <u>all</u> particulate so as to protect the pump cylinder surfaces from scoring and damage. Ultra-filters cost about $20,000 to purchase, and operating costs are also high. Pump repair is cost and time consuming adding to the overall cost of the decontamination project.

Shot or Grit Blasting

This technique uses high pressure air mixed with steel shot or abrasive grit directed at the work piece to abrade the surface and remove the contamination layer. An integral vacuum system connected to the blaster head collects the shot or grit for separation by cyclonic action. The heavier shot is recycled and reused, and the lighter particulate directed to a bag filter for disposal.

This technique is effective for the removal of paint and thin layers of contamination from steel or concrete surfaces. If the total area to be decontaminated is relatively small (a few hundred square feet) the volumetric amount of contaminated shot or grit wastes is high for the area being treated. The process is relatively slow and labor intensive because of the small size of the blaster head to needed to be able to effectively vacuum the spent shot back into the recovery hose. If the work is being performed in a high radiation area, the worker exposure from adjacent radioactive sources may be the controlling factor. If the contamination has penetrated more than

an eighth of an inch or more into the surface this process would not be cost effective for achieving free release of the surface.

Concrete Scarification

Concrete scarification (removal of one-quarter inch thick layers of the surface) is performed using a pneumatically operated multiple piston / bit head moved over the contaminated surface. The head removes up to one-quarter of an inch of surface per pass. The chips and dust generated in the process are captured within a vacuum shroud surrounding the head and directed to a HEPA vacuum system for disposal. The technique is very fast and effective in surface removal, and can remove up to about one inch of thickness by repeated passes. Beyond one inch thickness, alternative techniques would have to be used. The scarification process (also called scabbling) is ideally suited for concrete because most contamination rarely penetrates more than about one-half inch in depth into concrete. Accordingly, complete removal of contamination is possible permitting free release of the surface.

The process works well on floors because the chips remain within the shroud and can be readily vacuumed up into a disposal drum. However, vertical surfaces is more difficult because the heavier chips and dust tends to fall away from the shroud and further contaminating adjacent areas. Very high vacuum systems and a tight shroud help to minimize this problem, but on large projects the dust buildup in adjacent areas can become a problem.

Chemical Decontamination

The interior of process systems can be decontaminated using various pretreatment and chemical solvents to remove the contaminant or the corrosion layer that may contain the contaminant. Numerous solvents have been developed for specific contamination films on carbon and stainless steel surfaces with good overall success. The types of solvents are beyond the scope of this paper, but are readily available in the literature. The process generally consists of flushing a pretreatment chemical throughout the system to prepare the corrosion film for removal. A subsequent flush of a mild or aggressive acid or caustic chemical removes all or most of the corrosion layer. If the system or component is to be reused, a passivation flush may also be used to chemically neutralize the surface and provide a protective film.

These processes have performed well when the corrosion film was adequately characterized and the particular solvent tested both in the laboratory and in pilot testing on similar surfaces. When these two critical steps are not taken the results are usually disappointing because off-the-shelf processes or chemicals and concentrations are not always suitable under all conditions. The processes generally require controlled temperatures, pH levels, concentrations and flow rates to effectively remove the contamination film. Organic based solvents tend to break down at high temperatures into their constituent elements and lose their effectiveness. As the solvent action begins to dissolve the film, the solvent concentration changes and the dissolution rate diminishes. In components of the system where the solvent fluid velocity is reduced such as within tanks, system traps and piping deadlegs, the scrubbing action of the fluid to remove the dissolved corrosion film is great diminished. In these regions the dissolved contaminants may precipitate out of solution and redeposit on the surface. Not only does the solvent fail to perform its intended function, but may create a new "hot spot" of radioactive contamination which may be worse than the original condition that the solvent flush was supposed to correct.

The processing of the waste generated from chemical solvents is often overlooked as part of the cost of decontamination. Organic solvents can be removed demineralization after filtration of the particulate flushed along with the fluid stream. The demineralization deposits the dissolved contaminants on the demineralization beds and the resulting "clean" effluent stream disposed to an existing waste processing system for further treatment, or discharged as clean water if residual concentrations are within specific disposal limits. However, the spent demineralizer beds are now highly contaminated with the removed contamination from the process piping and must be disposed of in accordance with normal plant procedures. The waste volume of the packaged bed usually far exceeds the volume of radioactive waste removed from the process piping. This bed disposal cost is often not recognized as an additional cost of processing. Nor is the additional worker exposure incurred in disposing of these beds included in the estimated exposure involved in the decontamination program.

Many chemical solvent processes are touted as being capable of decontaminating to unrestricted use levels using single or multiple flushes. Such claims should be evaluated carefully and the source of data checked for its relevance to the specific process system to be decontaminated. As noted earlier, all solvent processes should be laboratory tested and pilot tested in the field before embarking on a major decontamination program. A few dollars invested on the front end may save thousands or even millions of dollars on the tail end.

CONCLUSIONS

Cleanup technologies are available today that are effective in achieving high decontamination factors for specific applications. The criteria to be met for unrestricted use of decontaminated materials must be clearly understood so that a realistic program objective can be adopted. Prior to embarking on any decontamination effort, a detailed radioactive contamination characterization should be made to identify the type and concentration of contamination present in the system or structure to be removed. An inadequate characterization will most surely lead to disappointing results in the decontamination process applied. The program planner must take a hard look at proposed decontamination techniques to determine if they are feasible for the task at hand, and should verify vendor claims by contacting previous users of their success or failures with the technique. The overall decontamination program costs, exposure and waste volume need to be evaluated along with the projected decontamination factor when assessing the cost effectiveness of proposed process. As expected, actual field experience is generally more reliable than laboratory tests or vendor claims. The program planner needs to ensure that the proposed technique will perform the intended task. Nevertheless, as waste disposal costs continue to increase at escalation rates far exceeding the national inflation rate, reliable decontamination techniques are needed to minimize the volume of waste going to controlled burial grounds.

Decontamination Technology for Decommissioning of Nuclear Facilities

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ABSTRACT

During the decommissioning of a nuclear reactor facility, appropriate decontamination before and after dismantling of the facility is required for each stage of the process. As part of the technology development for the Japan Power Demonstration Reactor (JPDR) decommissioning program, several new decontamination methods have been developed at JAERI.

This paper describes the decontamination methods for the decommissioning and discusses their application for the reuse of dismantled components on the basis of decontamination experience at JPDR.

INTRODUCTION

A nuclear reactor facility can be decommissioned in several ways: dismantling early after shutdown, dismantling after mothballing, or entombment. In Japan, early dismantling is necessary due to limited land resources for nuclear installations and, any site has to be renovated in preparation for the next nuclear installation.

During decommissioning, appropriate decontamination is required for each stage of the dismantling, not only to reduce occupational exposure but also to reduce decommissioning waste. In Japan, no large scale decontamination has been performed, except in the JPDR, but some small scale decontaminations have been executed. Therefore, data obtained from the JPDR decontamination activities, such as decontamination of systems, dismantled components, and concrete buildings, are valuable.

This paper describes the decontamination methods for decommissioning, including the decontamination results for JPDR, and the decontamination techniques for reuse of dismantled components.

CLASSIFICATION OF DECONTAMINATION TECHNIQUE

The decontamination process may be divided into three major parts, as shown in Table 1: decontamination before and after dismantling, and decontamination of the facility building. Each part is described below.

Decontamination Before Dismantling

Decontamination Method:

Decontamination before dismantling is used to reduce the dose rate in the working area to decrease occupational exposure. The process can be divided into two major parts: decontamination of the primary coolant system, and of the large components, such as the radwaste storage tank.

In general, decontamination before dismantling does not need a high decontamination factor (DF, the ratio of activity before to that after decontamination), because its main purpose is to reduce occupational exposure. (On the other hand, a high treatment efficiency of liquid waste is required, in addition to a large throughput, such as can be provided by a cartridge filter and ion exchange resin, because the object to be decontaminated has a large volume, and produces a large amount of secondary waste.)

The decontamination before dismantling can be carried out by the means of the following methods:

Chemical methods	Mechanical methods
- Reduction	- Abrasives
- Oxidation	- Hydro pressure
- Redox reaction	- Blasting

The method to be used in a particular case is selected according to the shape of the object and the characteristics of its contamination. In general, chemical methods tend to be applied to the system, while mechanical methods tend to be applied to the decontamination of the larger components.

Characteristic of CRUD, and the Effect of Chemical Decontamination:

The decontamination efficiency is affected by the amount of cromium (Cr) in the CRUD due to the CRUD's dissolution mechanism. Table 2 shows the representative system decontamination methods and conditions. Fig. 1 shows the examination of the decontamination methods used at JPDR, using sample specimens from the JPDR and two commercial power plants (PWR and BWR).

The test results indicated that reducing decontamination methods (No. 1 to 5 in Table 2) were not effective for high Cr-containing CRUD samples (more than 15% Cr), while a decontamination factor (DF) of 10 was obtained in the case of low Cr-containing CRUD samples (less than 12 % of Cr), as shown in Fig. I. Furthermore, the results also indicated that the

addition of an NP or AP pre-oxidation step to the reducing decontamination step further improved the decontamination efficiency for any Cr-containing CRUD (See No. 6 to 8 in Table 2 and Fig. 1).

Decontamination After Dismantling

Decontamination after dismantling intends not only to recycle the dismantled metal components, but also to reduce the volume of radioactive waste. Most metal components are pipes, valves, pumps and tanks consisting of stainless steel, carbon steel and aluminum. This decontamination is carried out by removing the contamination with a base metal from the surface during dissolution or abrasion processes, because the decontamination after dismantling requires a high DF. However, there is no decontamination method which can apply to all kinds of dismantled components, since the dismantled components have various shapes and materials, as mentioned above. A few decontamination methods, therefore, are needed to treat them.

Representative decontamination methods of dismantled components are as follows.

Electropolishing Decontamination Method:

An electropolishing decontamination technique has been developed [1]-[3] for polishing metal surfaces. In general, the object to be decontaminated is set as the anode in an electrolyte. Electric current is then supplied to the anode and a cathode, thus promoting the anodic dissolution of the metal surface material. Oxidation films and surface contamination, including the deposited CRUD, are removed in this surface dissolution process. On the other hand, when the object to be decontaminated is used as the cathode, the bubbling by hydrogen gas which appears on the object's surface also promotes the removal of the deposited material. Electropolishing decontamination is divided into a few methods, as shown in Table 3. Moreover, Table 4 gives the characteristics of the electrolyte: phosphoric acid, sulfuric acid and sodium sulfate (neutral salt).

This method is adequate for simple-shape components and is expected to attain a considerably high DF.

Chemical Immersion Decontamination Method:

In general, the chemical immersion decontamination method utilizes a chemical bath, and treats the object by two steps: pre-oxidation and reduction. This method uses many chemical solutions, depending upon the material of the objects. However, a system decontamination tends to become relatively complex, since this method uses two steps.

Also, there are several decontamination methods which need just one step of the redox reaction [4].

The chemical immersion decontamination method is adequate for complex shape components. Furthermore, simultaneous use of ultrasonic waves improves the decontamination efficiency, especially for objects with small crevices and holes.

Blasting Decontamination Method [5],[6]:

The blasting method decontaminates objects through the action of blasting abrasives, which are jetted from a nozzle with a fluid (air, water, or other solvents). Sand is an example of an abrasive, as is alumina, steel, boron carbide, silicon carbide and glass. This method uses a large amount of abrasives and fluid, so that separation of the abrasives and the fluid from contaminating slugs, and their reuse, are required to reduce the quantities of secondary waste.

A fairly high DF is expected of this method if carried out under appropriate conditions, which usually depends upon the object's material and the abrasives.

Decontamination of Buildings

A building is decontaminated not only to allow unrestricted use, but also to reduce the volume of concrete waste. The removal of only the concrete surfaces is sufficient for concrete decontamination, because contamination with Co-60 is limited to a few cm in depth.

Thermal stress methods such as a microwave irradiation, flame scarfing, and mechanical methods, such as milling cutter, scabbler, drill & spoller and blasting, are useful ways to decontaminate. The microwave irradiation method is especially good for decontaminating concrete buildings, because it does not make direct contact with the object being cleaned, and thus prevents the contamination from spreading. Also, it is not necessary to apply much mechanical force against the object (especially important for walls and ceilings).

DEMONSTRATION OF DECONTAMINATION METHOD IN JPDR

As part of the technology development for the JPDR decommissioning program at JAERI, a number of decontamination methods have been developed to be applied to the system before dismantling, to concrete surfaces, and to the dismantled JPDR components.

Several methods for system decontamination have already been demonstrated with the primary coolant system of the JPDR. In addition, the electropolishing decontamination technique for the dismantled components has started. And the decontamination systems for chemical immersion and blasting are planned to be manufactured in 1989 and 1990, respectively.

The results of the system decontamination and the electropolishing decontamination carried out in the JPDR are described below.

Decontamination Before Dismantling

Elemental Analysis of CRUD:

Results of elemental analysis of CRUD samples from the JPDR are shown in Table 5, in addition to samples from the commercial nuclear power plants (BWR and PWR).

The CRUD in the JPDR contains 15 to 25% Cr, while most CRUD in the BWRs contains less than 10% Cr.

Development of New System Decontamination Method [7]:

The redox decontamination method using sulfuric acid-cerium(IV) (the SC method, No. 9 in Table 2) and the flowing abrasive decontamination method (No. 10 in Table 2) have been developed at JAERI to overcome the Cr-rich CRUD problem as mentioned above. The SC method utilizing the redox reaction of Ce (IV) in a sulfuric acid solution is very effective in spite of the fact that it is a single-step decontamination method. The flowing abrasive decontamination method is independent of the chemical properties of CRUD, since this method utilizes only the mechanical attributes of abrasives. When grains of abrasives are forced into suspension in the water by circulation, the grains hit or rub the inner surfaces of pipes and components, thus removing the CRUD mechanically. The DF obtained by this method can be adjusted by the flow rate and the decontamination time. Figure 1 shows the test results after 12 hours of decontamination at a flow rate of 4.5 m/sec.

Demonstration of System Decontamination:

Four system decontamination methods have been demonstrated at the four parts of the JPDR primary coolant system, as shown in Fig. 2. The results by each decontamination method are shown in Table 6, along with the decontamination conditions.

(A) CAN-DECON Method

The CAN-DECON method was performed under normal conditions at the reactor's water clean-up line. This was done not only to get experience, but also to investigate unexpected problems that might arise during system decontamination. The maximum DF of 90 was obtained only at the regenerative heat exchanger, while the DF was 3 to 11 at the other parts of the line.

(B) Modified NP/NS-1

The modified NP/NS-1 decontamination method adds the NP pre-oxidation phase before the reducing decontamination phase. The addition of the pre-oxidation phase can considerably improve the decontamination efficiency even for high Cr-containing CRUD. In general, a 2-phase decontamination method needs a large amount of solution for each phase, and the solution is drained after each phase. For this reason, a liquid waste treatment system using a reverse osmosis (RO) module was introduced. Reuse of the water treated by the system can considerably reduce the total amount of water needed for this method. And, the liquid waste resulting from this method was a concentrated solution, which is less than 10% in volume of the total liquid used. The DF attained in this demonstration ranged from 90 to 740, and averaged about 500.

(C) Redox Decontamination Method

The redox decontamination method [7] using sulfuric acid and Cerium (IV) involves two processes: decontamination and liquid waste treatment. In the decontamination process [8], electrolytic regeneration of Ce (IV) maintains the constant concentration of Ce (IV), which controls the dissolution rate of the CRUD. The electrolytic regeneration keeps the ratio of Ce (IV) to Ce (III) less than 1. In the liquid waste treatment process, an electrodialysis is performed to collect metal ions and sulfuric acid from the liquid waste after their electrolytic reduction. The liquid

waste, containing few ions and little sulfuric acid, is further cleaned by an ion exchange resin. A DF of 300 to 1200 was attained, while the average DF was about 900 in the demonstration test of the JPDR primary coolant system [9].

(D) Flowing Abrasive Decontamination Method

The flowing abrasive decontamination method can be used at room temperature. Moreover, only an abrasive collector and a cartridge filter are required in the liquid waste treatment process, since most of the removed CRUD exists in the form of suspended solution. Therefore, a decontamination loop can be made very simple. The demonstration test of this method showed a DF of 200 to 1660, while average DF was about 1,100 [7].

Electropolishing Decontamination

Condition of Electropolishing and its Results:

From the results of the basic examination of electropolishing, optimum decontamination conditions for each electrolyte were determined to be as follows.

(A) Decontamination Condition for each Electrolyte

ELECTROLYTE	CONCENTRATION	CURRENT DENSITY	TEMPERATURE
phosphoric acid	80 wt%	0.2 A/cm^2	60°C
sulfuric acid	10 wt%	0.2 A/cm^2	60°C
Neutral salt (Sodium sulfate)	20 wt%	0.4 A/cm^2	60°C

(B) Optimum Cycle for Alternating Current Decontamination Method

Cathodic and an anodic electrolysis times are both 30 sec.

Electropolishing decontamination was performed under the above conditions using samples from JPDR and commercial power plants. Figure 3 shows the decontamination results for JPDR's primary coolant system pipes (SUS 304) for each decontamination method. The anodic decontamination method could rapidly decontaminate the samples in every electrolyte, and a DF of more than 5000 was obtained. The results also showed that the decontamination time required for the alternating current (a.c.) method was twice that of the anodic method, when sodium sulfate solution was used as an electrolyte. However, sufficient results were attained by every method after 20 minutes of decontamination. Figure 4 indicates the decontamination results for the RTD sample (SUS 316) of the commercial power plant (PWR) by each electrolyte. All samples contaminated with more than 10^5 Bq/cm^2 were perfectly decontaminated within 20 minutes by each method, and a DF of about 10,000 was attained.

The Factors of Electropolishing Decontamination Method:

If electropolishing can be applied for a long time, it can bring about complete decontamination. This method dissolves the contaminated metal surface. The decontamination

time, however, is one of the economical factors which affects the decontamination cost. In the case of decontaminating dismantled pipes or components, the time needed for electropolishing depends on the following factors: material, shape, extent of contamination, and degree of corrosion. These factors are described in detail below.

(A) Material

In electropolishing samples of stainless steel, a DF of more than 5000 was attained in a short-time decontamination, as shown in Fig. 3 and Fig. 4. Some carbon steel samples, however, could not be decontaminated, or else required very long decontamination times, even under the same conditions as stainless steel, Fig. 5. In general, carbon steel has thick CRUD layers on almost all surfaces, and corrodes non-uniformly. Though relatively good decontamination results were attained when 10 wt% sulfuric acid was used as an electrolyte, a fairly long decontamination time was required with phosphoric acid or sodium sulfate. However, the alternating current decontamination method was effective even if the electrolyte was sulfuric acid or sodium sulfate.

(B) Shape of Object to be Decontaminated

Electropolishing decontamination is very effective for simple components, such as pipes and boards, but it is sometimes difficult with complex shaped components, such as valves and pumps. The use of internal cathodes which fit to the shape of a component, however, can improve the decontamination efficiency. Figure 6 shows the decontamination results with the anodic method, using two internal cathodes for a valve casing. This valve casing was immersed in the electrolyte after setting two internal cathodes on sides A and B, Fig. 6. Contamination on side B was removed to below the lower limit of detection after 20 minutes of decontamination. Contamination remained in side A even after 30 minutes of decontamination. Figure 6 suggests that further decontamination was not effective for side A, and also that the anodic electrolysis method was not adequate for narrow or complex parts such as side A. Therefore, the electrolytic abrasive decontamination method was applied to remove the contamination that remained in side A. This method uses a rotary cathode made of a sponge with abrasives, located at the top of the pipe-shaft of a drill device. In addition, the rotary cathode is supplied with an electrolyte through the pipe-shaft. Consequently, decontamination can be achieved through both the mechanical rotation of the cathode and the electrochemical function of electropolishing. Figure 6 shows the results of the decontamination by this method, indicating that it could remove the remaining contamination in under two minutes.

(C) Effect of a Weld

Figure 7 shows the decontamination results of electropolishing for pipes with welded parts. Welding changes the characteristic of the base metal, thus leading to differences in thickness, characteristics, and form of the CRUD. For this reason, decontamination by electropolishing tends to become slightly difficult.

(D) Effect of a Small Hole or Crevice

Electropolishing decontamination of components with small holes or crevices is difficult, because the electric current density becomes non-uniform. The internal cathodes or the

electrolytic-abrasive decontamination method is recommended, following decontamination by the immersion method.

(E) Effect of Edges Cut by the Thermal Method

An edge cut by the thermal method (such as plasma-arc), has contamination in the base metal, which melted at the moment of cutting. In addition, the thermal cutting method spatters melted metal, including contamination, near the cutting point. For this reason, the contamination remains at the melting parts and the spatters, and a high DF can not be attained.

Decontamination of a Concrete Building

The microwave irradiation method had been developed in the JPDR to demolish concrete surfaces to a depth of a few centimeters [10], [11]. An examination was performed using a prototype microwave irradiation machine. In addition, existing mechanical concrete decontamination methods such as milling cutter, scabbler, blasting, drill & spoller, and hand-breaker were also examined. The milling cutter [12] and the scabbler were adequate for the removal of concrete surfaces to a depth of 3 to 5 mm. The microwave irradiation method is adequate in removing concrete surfaces to a depth of a few cm. This is important for walls and ceilings, because the method does not involve applying force against the object.

A demonstration machine has been manufactured, and a test of cold samples has been performed. This machine is planned to be applied to the JPDR concrete building in the controlled area from 1990 to 1991.

DISCUSSION

Demonstrations of decontamination methods at JPDR showed that the dismantled simple-shape components contaminated at about 3.7×10^2 Bq/cm^2 (1.0×10^2 μCi/cm^2) could be decontaminated to below the lower limit of detection. In addition, examination of the system and of the electropolishing decontamination of commercial power plant samples indicated that the components of a commercial power plant could also be decontaminated to below the lower limit of detection.

However, measuring decontaminated components becomes very hard when they are decontaminated to a very low level. Because the measurement area has a relatively high dose rate, in general, a shield is required to enclose the components. In addition, most components to be decontaminated such as pipes, valves, and pump casings range from a few 10 kg to a few 100 kg in weight, but their plane surfaces are not large. A detector having a large detection-area, therefore, can not be used with them.

At JPDR, a GM survey meter with a detection area of 20 cm^2 is used mainly for measuring decontaminated components. A very long time period is needed for these measurements. For example, a GM survey meter used to measure a decontaminated pipe 400 mm in diameter and 700 mm in length requires 100 minutes under the following condition:

Background : 80 cpm
Measurement time for each point : 10 sec
Detection area of the detector : 20 cm^2
Time for change the position of the detector : 5 sec/point
Low limit of detection : 1.4×10^{-1} Bq/cm^2
(3.88×10^{-6} $\mu Ci/cm^2$)
Number of workers required : 1 person

In this case, the electropolishing decontamination was finished within 30 minutes (during which the electric current was supplied), and no contamination could be detected afterwards.

Most components produced by decommissioning can be completely decontaminated if factors such as cost, occupational exposure, and secondary waste can be ignored. However, these factors are not negligible when the decontamination for reuse is undertaken.

CONCLUSION

The results of the system and electropolishing decontamination studies indicate complete decontamination for reuse is possible.

For system decontamination, a DF of about 50 provides sufficient reduction of the dose rate (less than 30 mR/hr in the working area). However, the reuse or recycling of dismantled components requires a higher DF of about 200. The results of the system decontamination showed that an average DF of more than 500 could be attained if both the decontamination method and conditions were appropriately selected according to the Cr content in the CRUD.

For the electropolishing decontamination, it was demonstrated that a high DF (more than 10^4) was obtained by the anodic electrolysis method for both stainless and carbon steal simple-shaped components. Also, the complex-shaped components could be decontaminated either by anodic electrolysis with an internal cathode or by electrolytic-abrasive decontamination methods.

However, cost and occupational exposure are also important considerations in selecting the decontamination method for reuse. The cost of contamination measurements of decontaminated components, in particular, becomes important in the case of reuse.

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Table 1 Decontamination for Decommissioning

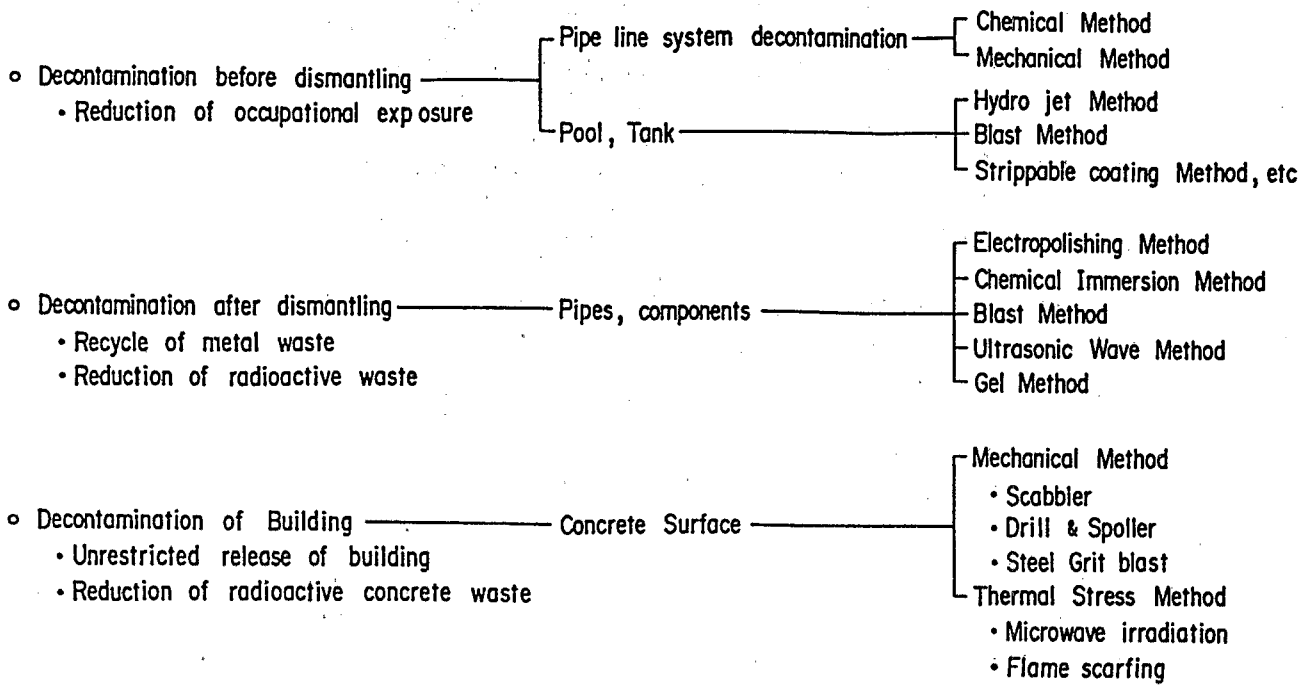


Table 2 Decontamination Method tested in Basic Examination

No.	Name of Decon.	Decontamination condition					Classification	Memo
		Decontamination Agents		Conc.	Temp.(°C)	Time (hr)		
1	NS-1 (dil)	NS-1		0.7 wt%	120	24	Reducing Decontamination	
2	CAN-DECON	LND-101A		0.1 wt%	120	24		
3	EBARA-DECON	ED-40 (Modified)		2.0 wt%	120	24		
4	KURI-DECON	KD-203		0.1 wt%	120	24		
5	NS-1	NS-1 (conc.)		7.0 wt%	120	100		
6	NP/NS-1 (dil)	NP	$KMnO_4$ HNO_3	1 g/ℓ 5 g/ℓ	120	6	Pre-oxidation + Reducing Decontamination	Corrosive
		Reduction	Oxalic acid	1.42 g/ℓ	60	0.75		
		Decon.	NS-1	0.7 wt%	120	24		Corrosive
7	NP/CAN-DECON	NP	$KMnO_4$ HNO_3	1 g/ℓ 5 g/ℓ	120	6		
		Reduction	Oxalic acid	1.42 g/ℓ	60	0.75		
		Decon	LND-101A	0.1 wt%	120	24		Corrosive
8	AP/Citrox	AP	$KMnO_4$ NaOH	32 g/ℓ 105 g/ℓ	105	4		
		Ci	Oxalic acid Ammonium Citrate, Dibasic Iron Nitrate Diethylthiourea	25 g/ℓ 50 g/ℓ 2 g/ℓ 1 g/ℓ	85	48		
9	SC Method	Sulfuric acid Cerium(IV) Sulfate Ce (IV)		14.5 g/ℓ 2.06 g/ℓ (4.8 mmol)	80	48	Redox decontamination	Corrosive
10	Flowing Abrasive	B_4C (Abrasives)		20 wt%	Room temp.	12	Mechanical Method	

Table 3 Electropolishing Decontamination

- Immersion System
  - Anodic Electrolysis Decontamination Method
  - Cathodic Electrolysis Decontamination Method
  - Alternating Current Electrolysis Decontamination Method
- Electrolytic - Abrasive Decontamination System
- In - Situ Technique
  - Pumped Stream Method
  - Contact Method

Table 4 Characteristics of Electrolyte

Electrolyte	Surface finishing	Solution life	Carrying-out	Rinse	Purpose
Phosphoric acid 70 ~ 80 %	bright surface	50 ~ 70g/ℓ (metal ion conc.)	Large (1 mℓ/cm²)	Difficult	Reuse
Sulfuric acid 5 ~ 20 %	rough etched surface	20 ~ 30g/ℓ (metal ion conc.)	Small (0.5mℓ/cm²)	Easy	Recycle
Neutral salt 20 %	rough etched surface	Long	Small (<0.5mℓ/cm²)	Easy	Recycle

Table 5 Chemical Composition and Activity of CRUD

Type of Reactor	Part	Amount of CRUD	Composition (%)				Radioactivity
			Fe	Cr	Ni	Co	
Commercial Power Plant (PWR)	RTD, 2"ϕ	690 $\mu g/cm^2$	59.7	22.1	16.3	0.29	5.8 $\mu Ci/cm^2$ (2.1×10^5 Bq/cm^2)
Commercial Power Plant (BWR)	Reactor Water Clean-up Line	990 "	75.0	10.2	12.8	0.32	4.8 $\mu Ci/cm^2$ (1.8×10^5 Bq/cm^2)
JPDR	Reactor Water Clean-up Line	714 "	47.3	24.4	27.9	0.01	0.062 $\mu Ci/cm^2$ (2.3×10^3 Bq/cm^2)
JPDR	Recirculation Line	400 "	64.4	12.2	21.2	0.34	0.025 $\mu Ci/cm^2$ (9.3×10^2 Bq/cm^2)

Table 6 Condition & Results of System Decontamination Methods applied to the JPDR Primary Coolant System

Decontamination method		CAN-DECON	NP / NS-1 modified		Sulfuric acid-Ce(IV) (SC Method)	Flowing Abrasive
		Reducing Decon.	Pre-oxidation	Reducing Decon.	Redox Decon.	Mechanical Decon.
Condition	Regents	LND-101A	NP (HNO_3 / $KMnO$)	NS-1	H_2SO_4 + Ce^{4+}	B_4C Particles
	Concentration	0.1 wt %	0.6 wt % (0.5 wt % / 0.1 wt %)	0.7 wt %	0.25 M 2mM	20 wt %
	Temperature	~120 °C	~120 °C	~120 °C	70 ~ 80 °C	Room Temp.
	Decontamination Time	24 hr	6 hr	24 hr	106 hr	35 hr
	Flow rate	——	——	——	——	4.8 ~ 6.7 m/sec.
Regeneration of Decon. solution		Cation resin	non		Electrolytic reduction	non
Treatment of Waste Solution		Mixed bed resin	Reverse Osmosis		Electrodialysis Mixed bed resin	Cartridge Filter
Result	Decontamination Factor (DF)	3 ~ 90 av. 9	90 ~ 740 av. 520		300 ~ 1200 av. 900	200 ~ 1660 av. 1100
	Form of Waste	Cartridge Filter Resin	Condensed Waste Solution (15% of system volume) RO Module		Condensed Waste Solution Regenerated Sulfuric Acid Ion Exchange Membranes Cartridge Filter Resin	Abrasives Sludge Cartridge Filter

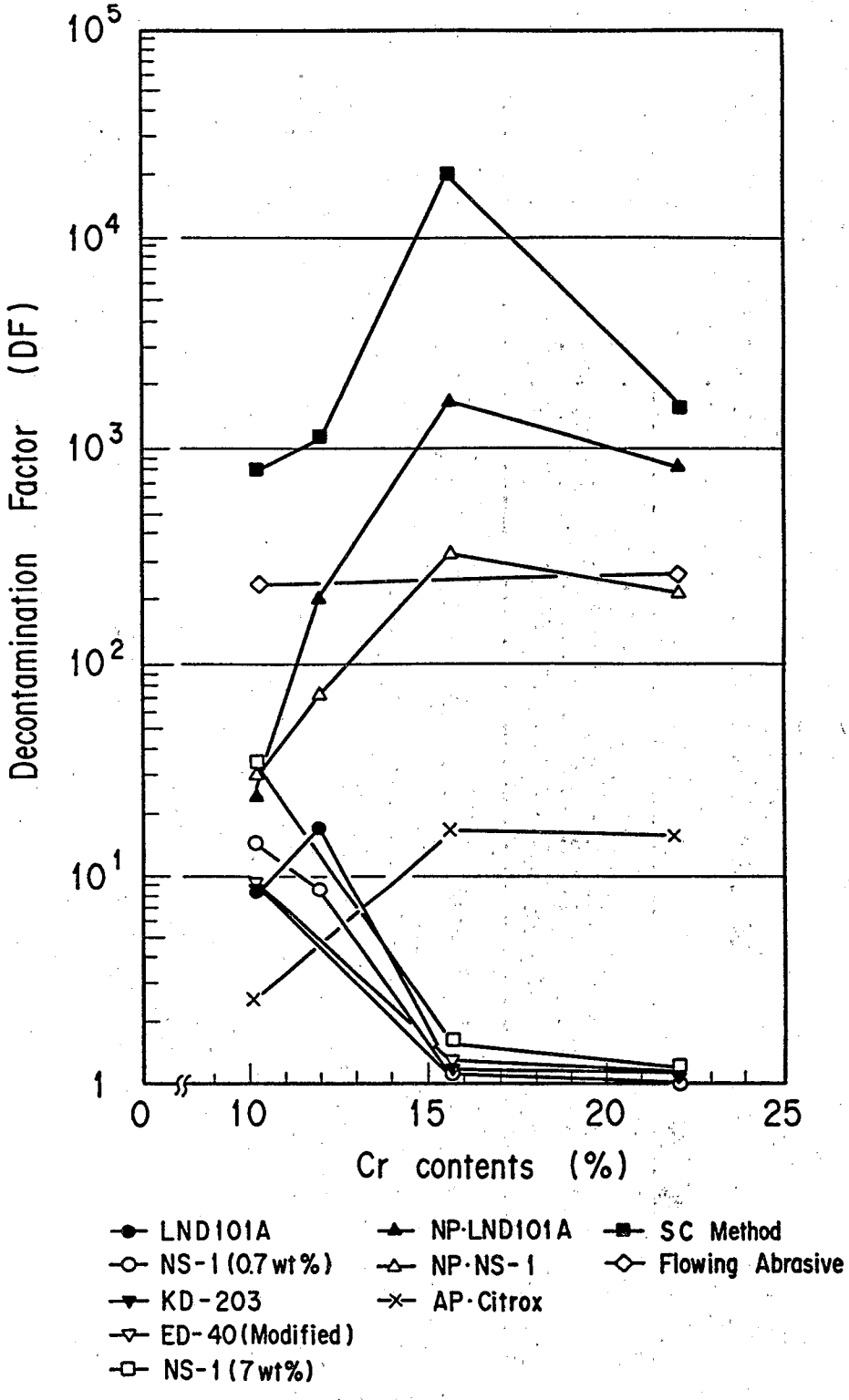


Fig. 1 Relation between Cr-contents and Decontamination Factor in case of Basic Examination by Various Decontamination Methods

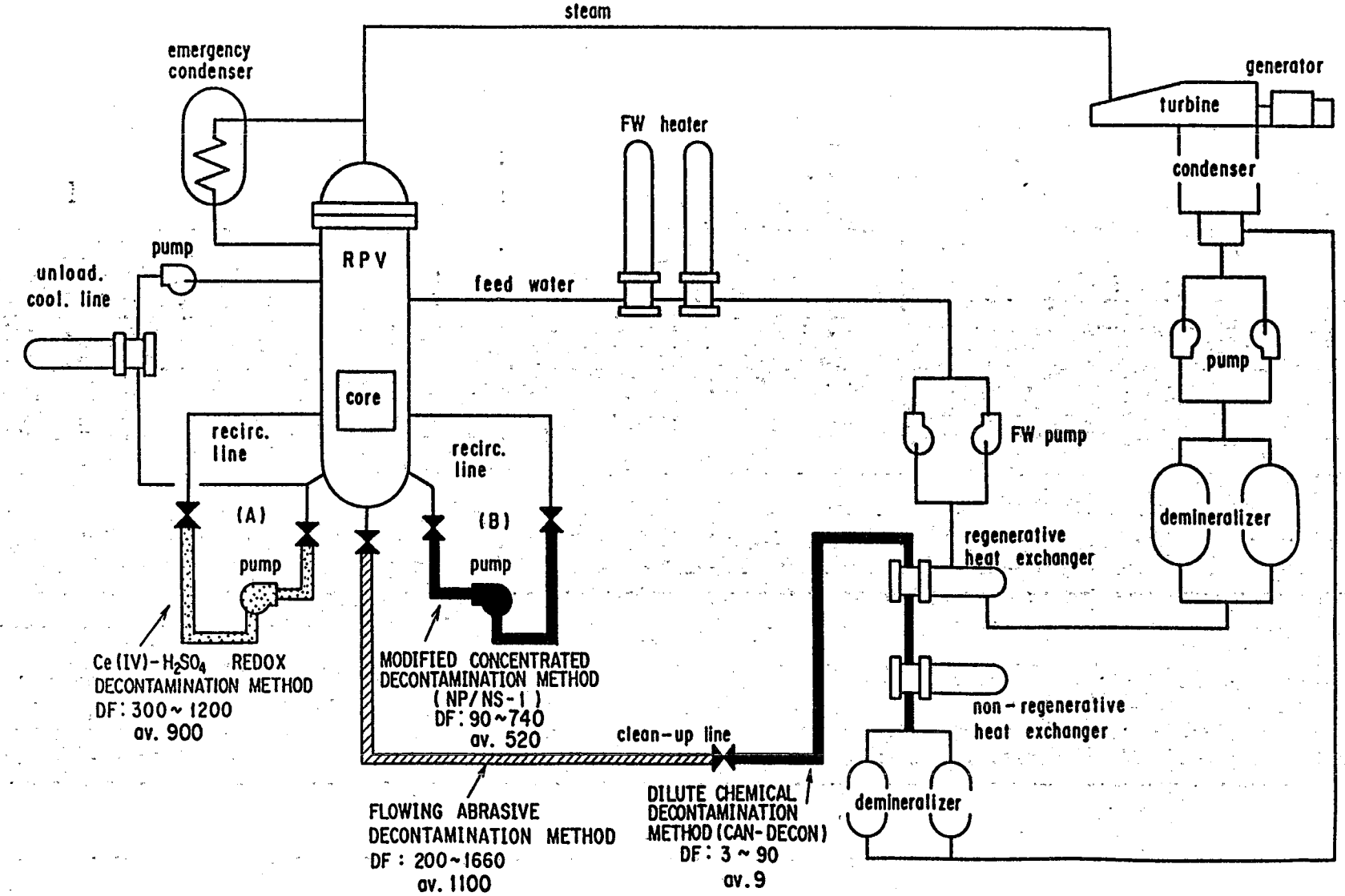


Fig. 2 Decontamination Methods applied to the JPDR Pimary Coolant System

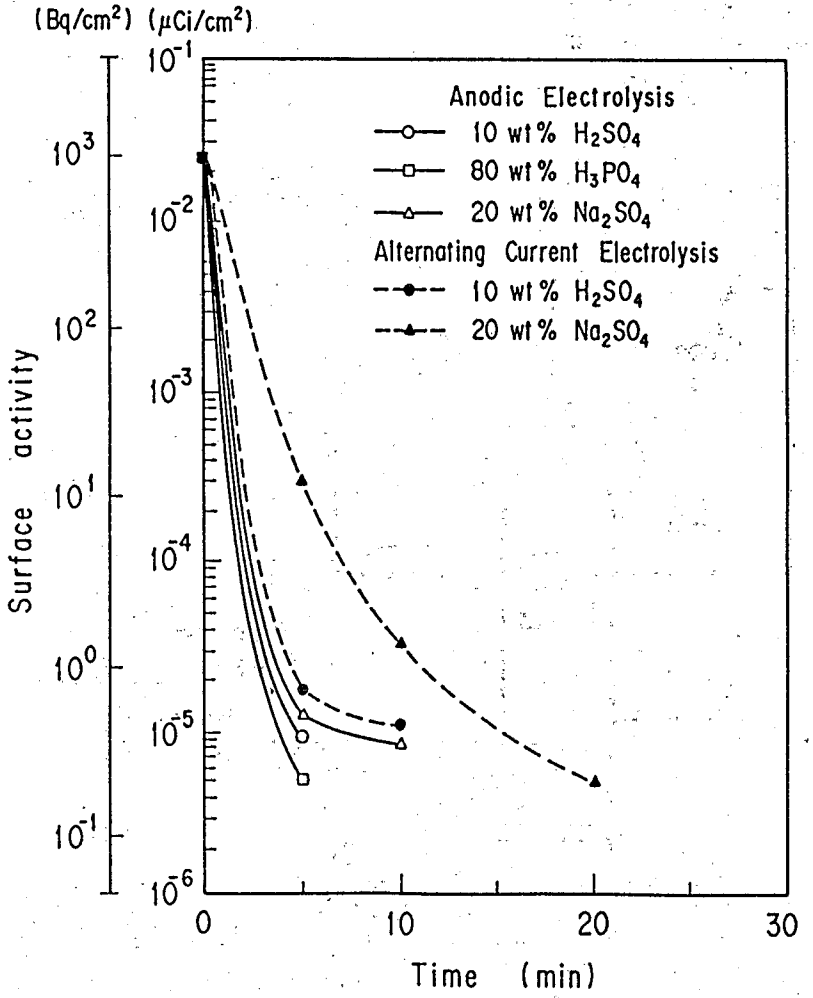
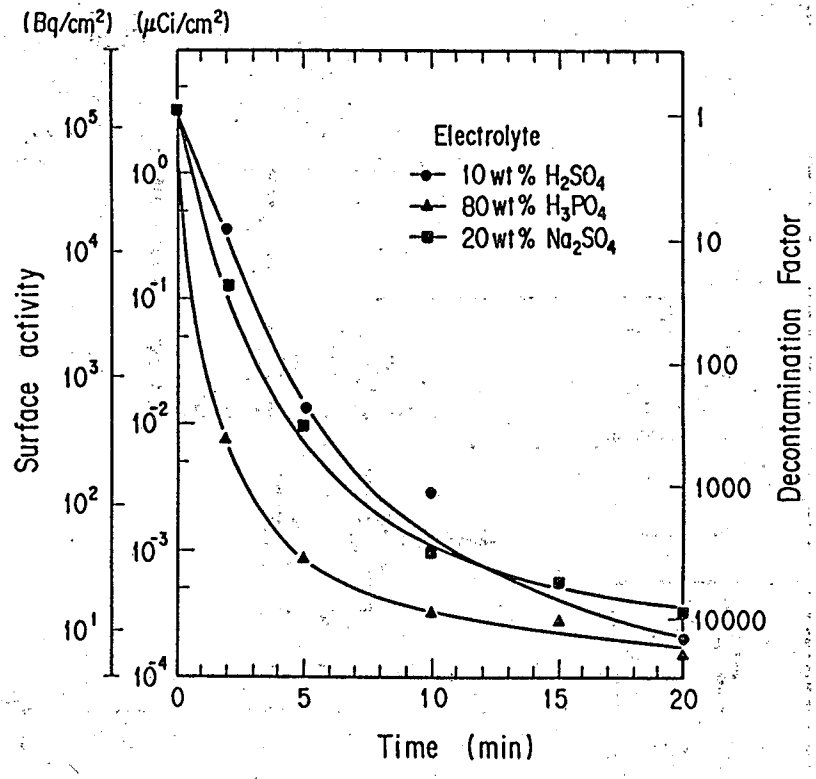


Fig. 3 Results of Basic Examination of Electropolishing using Stainless Steel Samples of the JPDR



Sample	: RTD of Commercial power plant (PWR)
Decon. Method	: Anodic Electrolysis with Immersion System
Material	: SUS-316
Temp. of Electrolyte	: 60℃
Current Density	: 0.2A/cm²

Fig. 4 Results of basic Examination of Electropolishing with PWRs Samples

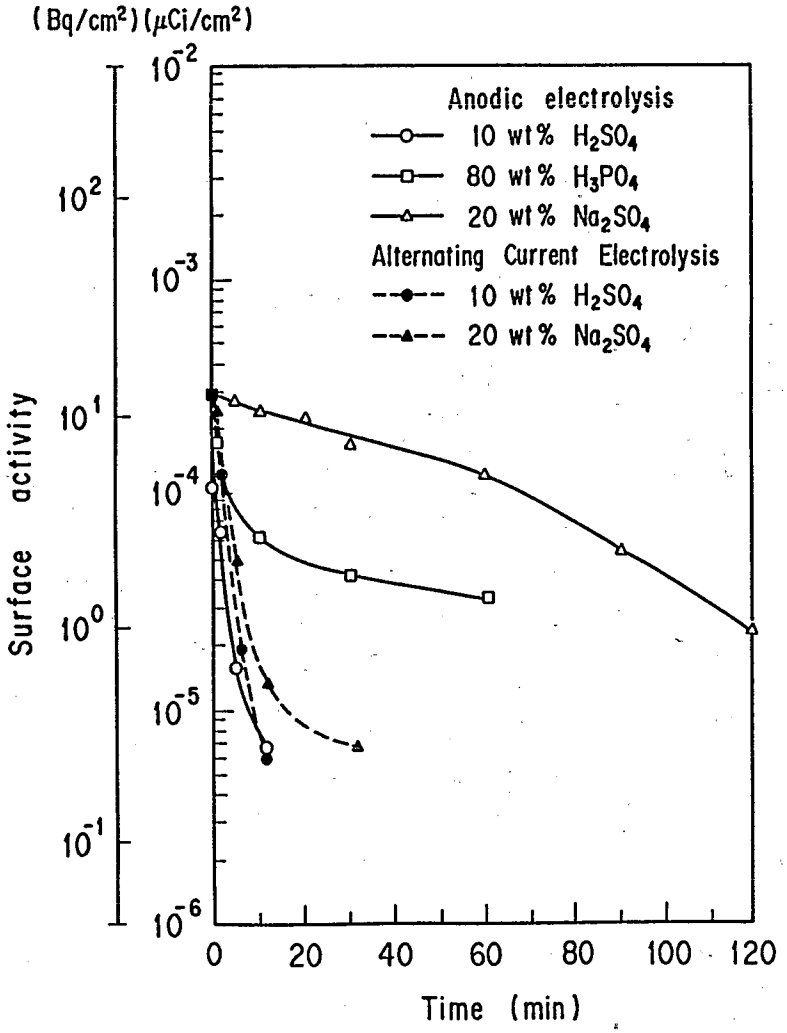
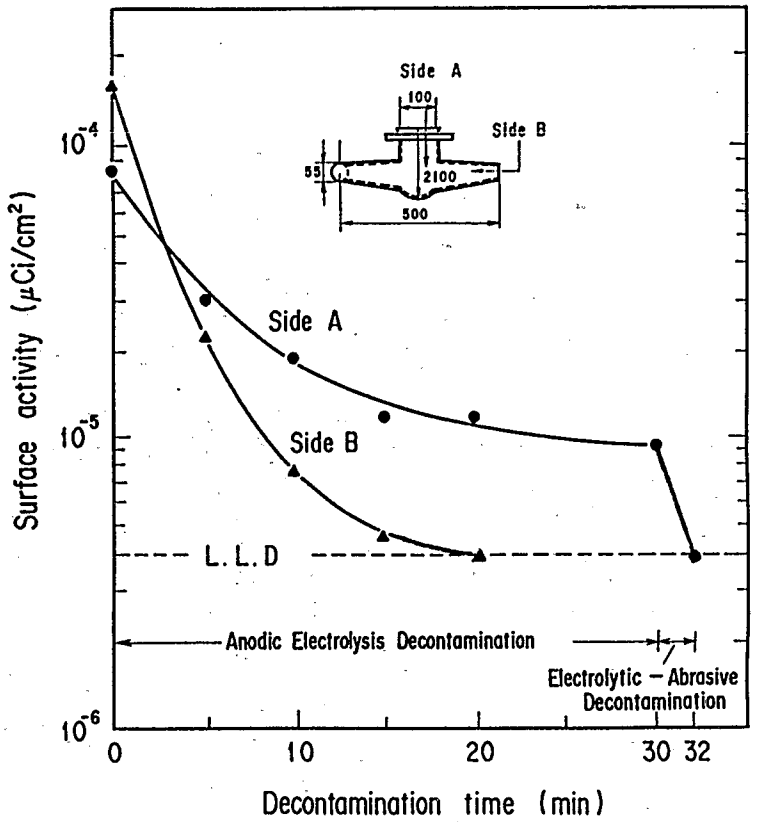
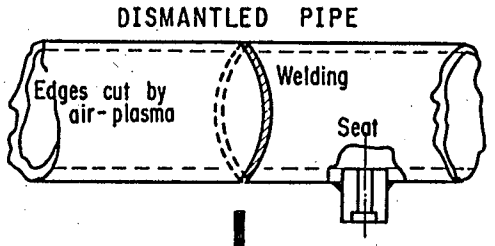


Fig. 5 Results of Basic Examination of Electropolishing using Carbon Steel Samples of the JPDR



Anodic Electrolysis Decontamination with Immersion System		Electrolytic – Abrasive Decontamination	
Electrolyte	: 10 wt % H_2SO_4	Electrolyte	: 20 wt % Na_2NO_3
Material	: SUS	Material	: SUS
Current density	: 0.2 A/cm²	Current density	: 5 A/cm²
Temp. of electrolyte	: 80°C	Temp. of electrolyte	: Room Temp.

Fig. 6 Relation between Surface Activity of Complex Sample and Decontamination Time



Decontamination condition

Electrolyte	: 80 wt% phosphoric acid
Temp. of electrolyte	: 60°C
Current density	: 0.2 A/cm^2
Decontamination time	: 30 min
Sample	: SUS 304
L.L.D	: 3×10^{-6} μCi/cm^2 (1.1×10^{-1} Bq/cm^2)

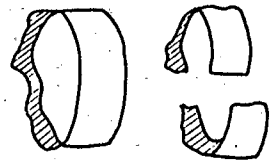
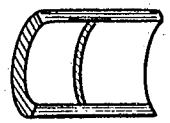
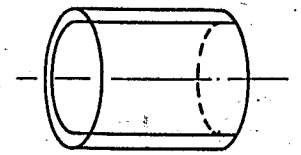
Edges cut by air-plasma	Simple pipes	Pipes including welding	Pipes with seat	Pipes with spatters
Thermal cutting method put contamination into metal. Complete decontamination is not possible.  Ultra sonic decontamination is not effective.	No contamination was detected after decontamination.	Contamination remains on welding parts. It can be removed by mechanical decontamination method.  Ultra sonic decontamination is not effective.	Contamination remains on inner surface of hole and around the seat.	Pipe cut by thermal method has spatters near its edges. Spatters remain after decontamination. Contamination was detected at the spatters.  Spatters can be removed by mechanical decontamination method. No contamination detected after the decontamination.
DF 50 ~ 400	DF 600 ~ 1300	DF 700 ~ 900	DF 20 ~ 500	DF 600~1300 (After mechanical decontamination.)

Fig. 7 Results of Electroplishing Decontamination of Various Parts of Pipes

Low-Level Radioactivity Measurement Methods for Reusing or Recycling

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ABSTRACT

When reusing facilities and land, or recycling materials, radioactivity surveys and measurements are indispensable to make the program effective and to assure radiologic safety.

We discuss the procedures and techniques of radioactivity measurement needed to confirm the contamination levels for reuse or recycling. These procedures can be divided into three phases: planning, decontamination, and reuse or recycling. Radioactivity measurements should be made in each phase: In the first, pre-investigating the contaminated spots or materials; in the second, checking the contamination levels and classifying the materials; and in the third, verifying the contamination levels and confirming the radiologic safety. Some methods and instruments appropriate for each phase are proposed, and the measurements (including minimum detectable limits) are discussed. New kinds of automated systems developed at JAERI for verifying low level radioactive wastes are presented. One of these has a minimum detectable limit of 10 mBq/g for Co-60 and Cs-137.

We shall illustrate the procedures and measurement techniques with three examples: the reuse of research reactor facilities, the reuse of the radiochemical laboratory, and the dismantling of a power reactor. Our experience so far has shown the effectiveness of the survey and measurement techniques and of the automated measurement system for 200 liter drums.

INTRODUCTION

When reusing facilities and land, or when recycling materials, radioactivity surveys and measurements are needed to make the program effective and to assure radiologic safety. Since the objects of the surveys and measurements vary in amount, form and contamination level, appropriate methods must be applied to each case. This may also require an apparatus or a measurement system designed exclusively for a particular case.

We will discuss the procedures and techniques of radioactivity measurement needed for reuse or recycling. This paper gives three application examples: reuse of research reactor facilities, reuse of a radiochemical laboratory, and dismantlement of a power demonstration reactor.

PROCEDURES OF RADIOACTIVITY MEASUREMENT FOR REUSING OR RECYCLING

The major objects of reuse and recycling are buildings and land dedicated to processing or treatment of radioactive materials, and structured materials used in the controlled area. The major items requiring measurement are floors, walls and ceilings of a facility for reuse, and steel plates and pipes removed from the facility for recycling.

The process of reuse or recycling of a facility consists of three phases: planning, decontamination, and reuse or recycling. The radioactivity measurement procedure is divided into three corresponding phases, as shown in Figure 1. The measurements in the first phase are needed to make a reuse or recycling plan. The measurements in the second phase serve to check the contamination levels of the buildings and the land, and to classify the structured materials removed from the controlled area. The measurements in the third phase verify the contamination level of the building, land, and the removed materials, and confirm their radiological safety before reuse and recycling.

MEASUREMENT METHODS

Principal long-lived nuclides to be measured are commonly used nuclides such as Co-60, Sr-90, Y-90 and Cs-137, and activation nuclides such as Cs-134, Eu-152 and Eu-154.

Various radioactivity measurement methods have been proposed for reuse and recycling. It is important to choose a method or a combination of methods based upon their merits and the contamination situation at hand. Measurement should be applied to large amounts of material.

There are four classes of methods:

A. Survey on the spot, using portable contamination meters.

A portable contamination meter offers easy and flexible operation. The instrument is inexpensive, but requires substantial labor input. Typical measurement instruments for surface contamination are shown in Table 1. A gas flow proportion counter with a large thin window has a minimum detectable limit (MDL) less than 0.1 Bq/cm^2 in a short counting time for alpha/beta-ray.

This method is applicable for all three phases discussed in the preceding section.

B. Measurement of small specimens sampled at the selected location using sophisticated instruments.

Typical instruments are gamma-ray spectrometers, low background alpha/beta counters, and liquid scintillation counters. Such instruments provide detailed analysis with high precision, but they are expensive and require operators. Typical radioactivity measurement instruments and their performances for sampled specimens are shown in Table 2, which also includes information of ordinary survey meters used for measuring radioactive steel [1].

It is generally possible to lower MDL by extending the counting time, or by increasing the amounts or concentrations of specimens. However, the number and the kind of specimens and total costs should be considered when making a decision.

This method is applicable for all phases, and for measuring contamination of either structured materials or removed materials.

C. Measurement of large amounts of material using an automatic measurement instrument.

Such a system measures mainly gamma-rays from removed materials in drum-sized containers, and is better when custom designed for this purpose. The systems must be installed near the working area. This is expensive, but it allows a large amount of material to be checked effectively and quickly.

This method is applicable in the second and third phases.

Some systems for which this method has been developed are as follows:

- A measurement system for active neutron assays of TRU waste in 200 liter drums (MDL : >100 Bq/g for Pu-239) [2]
- An automatic measurement system using 4 Nai (Tl) gamma-ray spectrometers for waste canned in 200 liter drums (MDL : 40 mBq/g for Cs-137)
- An automatic measurement system using 2 Ge gamma-ray spectrometers for waste canned in 200 liter drums (MDL : 10 mBq/g for Cs-137, Co-60) [3], [4]. (described later)

D. Consecutive monitoring of materials conveyed continuously using an automated instrument with a warning function.

These systems should also be custom designed. The method has the same features as the third one, except for its inferiority in analyzing capability. Similar systems have been developed for checking articles from controlled areas. A typical one is the contamination inspection monitor, which has MDL of 0.1 Bq/cm^2 for Co-60.

In the planning stage, the first phase, it is important to make a careful investigation of the facility's history and the characteristics of the contamination spot, e.g. major radionuclides, contamination level and expanse, with the methods of A and B presented above.

As to the measurement technique, it is necessary to pay attention to the radionuclides resulting from elsewhere, such as nuclear tests or Nature, which could potentially interfere with the analysis.

MEASUREMENT EXPERIENCES FOR REUSE OF FACILITIES AT JAERI

This section describes three cases in which the policy discussed in the preceding two sections was applied. Specific measurements were made for reuse programs of a research

reactor facility, a radiochemical laboratory, and the land occupied by a power reactor demonstration facility.

1) JRR-3 facilities (Japan Research Reactor No.3)

JRR-3 is a thermal-heavy water reactor fueled with natural uranium and operated for about 20 years. Investigation in the planning stage verified considerable H-3 contamination of the whole reactor containment area, and surface spots of Cs-137 contamination scattered on the floor.

H-3 was found to permeate deep into the concrete. H-3 activities of bored samples were measured with a liquid scintillation counter. The counter has a MDL of 0.4 Bq/g. The floor surface contamination, on the other hand, was surveyed with a gas flow proportion counter-type contamination meter with a large thin window (GPCM). The instrument has a MDL of 10 mBq/cm^2. Samples of the removed materials were analyzed with a Ge gamma-ray spectrometry system. The system has a MDL of 2 mBq/g.

These three methods were applied to investigate in the first phase, and to classify materials in the second phase.

The reactor containment was released from control after appropriate cleanup, and a new research reactor has been constructed within the containment. Contaminated materials were classified into two categories: (1) materials contaminated with Cs-137 or with H-3 over 37 Bq/g, and (2) materials contaminated with H-3 below 37 Bq/g. The former was disposed of as low-level solid waste, and the latter has been under control in a storage facility.

2) Radiochemical laboratory

Research Laboratory Building No. 1 included many radiochemical facilities in which unsealed radioactive materials have been treated for more than 20 years. It was remodeled for reuse as an ordinary office building after cleanup.

A careful investigation in the planning stage, verified about 330 scattered spots of surface contamination with C-14, Co-60, Ru-106, Cs-137, U and Th, with maximum concentration of 400 Bq/cm^2.

The surface contamination of the floor and equipment was surveyed with a GPCM. Samples of scrapped materials, such as concrete fragments and pipe pieces, were analyzed with a Ge gamma-ray spectrometry system.

The facilities were released from control after a series of elaborate cleanups and have been reused as an ordinary office building. The scrapped waste materials were categorized into two groups: (1) contaminated materials, and (2) those below the minimum detectable limit, and sure to be uncontaminated. The former was disposed of as low-level solid wastes, and the latter as ordinary industrial waste within the site.

3) JPDR facilities (Japan Power Demonstration Reactor)

JPDR is a BWR-type demonstration reactor that has been operated for 17,000 hours. The reactor facilities have been dismantled to allow study of the technology, and to reuse the site. Heavily contaminated materials, such as spent fuel assemblies and interior parts of the reactor itself, have been removed from the facilities.

A comprehensive investigation, in the planning stage, made clear the radiological impact and led to a concrete dismantling program.

The contamination could be divided into three groups: (1) activated materials in the biological shield around the core, (2) fission products in the piping, and (3) fission products on the surface leaked from the piping.

Measurements of density and nuclide contamination have been made with a GPCM and Ge gamma-ray spectrometry system, respectively.

An automated measurement system was developed exclusively to confirm the extremely low-level waste contained in the 200 liter drums. The system has a computerized gamma-ray spectrometer with two scanning Ge detectors, and it can measure the total radioactivity in a drum in about 10 minutes. The system has a MDL of 10 mBq/g for Co-60 and Cs-137. The total activity in a drum measured with the system is in agreement with that obtained by measurements of sampled specimens.

CONCLUSION

We have presented the radioactivity surveys and measurement techniques in connection with reuse of facilities and land, or recycling of removed materials.

We have applied our low-level radioactivity measurement methods to the reuse of a research reactor facility, a radiochemical laboratory, and the dismantling of the power demonstration reactor in JAERI. The gas flow proportion counter-type contamination meter with a large thin window (GPCM) for checking surface contamination and the Ge gamma-ray spectrometer for measuring sampled specimens have proven to be excellent in sensitivity, response, cost and measurements. And the automated measurement system for 200 liter drums has been effective for large quantities of material.

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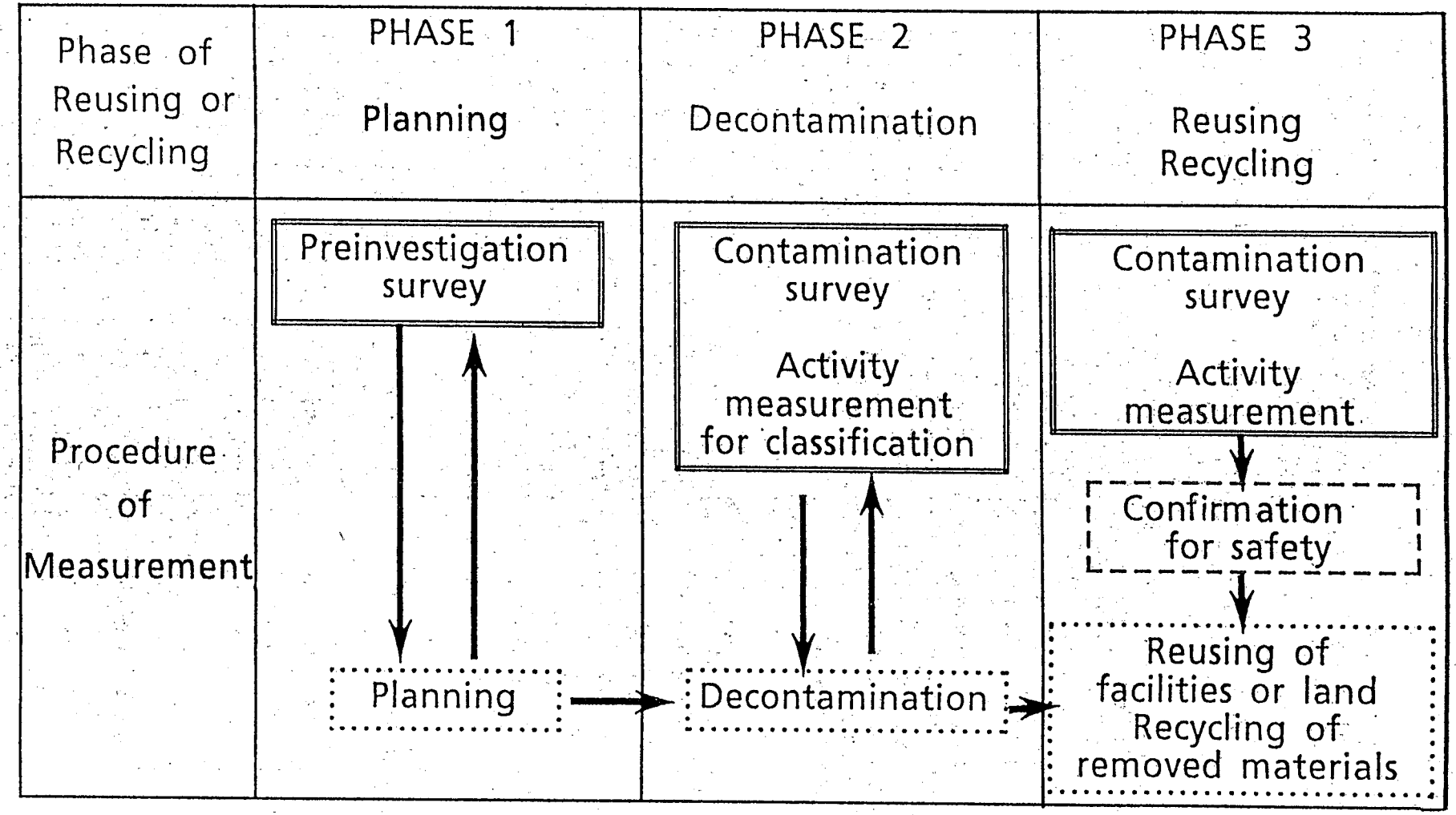

Figure 1 Procedure of Radioacitivity Measurement for Reusing or Recycling

Table 1 Typical Measurement Instruments for Surface Contamination

Measured item	Sample shape	Instrument	Detector Type	Detector Size (cm^2)	Radiation & typical nuclides	Time constant or count time*(s)	Minimum detectable limit (Bq/cm^2)
Floor Wall	Flat	GM survey meter	GM tube	20	$\beta(\gamma)$ ^{60}Co, ^{137}Cs	10	0.4
		Gas flow proportional contami. meter	Gas flow proport.	100	α U	10	0.01
					β ^{60}Co, ^{137}Cs	10	0.05
		Plastic scintillation contami. meter	Plastic scintil-lation	50	β ^{60}Co, ^{137}Cs	10	0.08
Floor	Flat	Floor contami. monitor	Gas flow proport.	1800	β U	10*	0.04
					^{60}Co		0.1
Article, Steel, etc.	Plate Box	Contamination inspection monitor	Plastic scintil-lation	5250	β U	600*	0.04
					^{60}Co	600*	0.1

Table 2 Typical Radioactivity Measurement Instruments for Sampled Specimen

Measured item	Sample amount (g)	Instrument	Detector Type	Detector Size (cm^2)	Radiation & typical nuclides	Time constant or count time*(s)	Minimum detectable limit (Bq/g)
Concrete Steel	10	GM counter	GM tube	20	$\beta(\gamma)$ ^{60}Co, ^{137}Cs	600*	0.4
	1	Low back ground α/β counter	Gas flow proportional	20	α U, Th	600*	0.005
					β ^{60}Co, ^{137}Cs	600*	0.01
	1	SSD α-ray spectrometer	Si surface barrier	10	α U, Th	10000*	0.04
	500	Ge γ-ray spectrometer	HpGe	150 (cm^3)	γ ^{60}Co, ^{137}Cs ^{134}Cs, ^{152}Eu	500*	0.002 0.003
Concrete	1	Liquid scinti. counter	LSC	20 (cm^3)	β ^{3}H, ^{14}C	600*	0.4
Steel Disk	30cm (D)x 2.7cm (T)	NaI survey meter	NaI Scinti.	2.5cm Dx 2.5cm L	γ ^{59}Fe, ^{54}Mn	10	0.1
SUS Pipe	2.5cm(D) x 50cm (L)	NaI survey meter	NaI Scinti.	2.5cm Dx 2.5cm L	γ ^{51}Cr	10	0.4

Disposal Capacity and Projected Waste Volumes Within the Low-Level Radioactive Waste Compacts

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ABSTRACT

Waste volume and activity projection analysis have been performed for the Southwestern and Central States projects. A detailed discussion of the data and projections from fuel cycle generated waste, medical wastes, academic, government and non-medical industry waste is presented. The effects of nuclear power plant decontamination, volume reduction technology, U.S. NRC/U.S. EPA BRC decision, and mixed waste production are discussed. The projected waste volumes and activity projections are compared with present disposal capacity.

In the late 1970s, the governors of Washington, South Carolina and Nevada declared that they would close their facilities to out-of-state LLW if swift progress were not made in developing new disposal sites in other regions of the country. In response to the concerns of these states, Congress in 1980 passed the Low-Level Radioactive Waste Policy Act. This Act required each state to be responsible for providing disposal capacity for LLW generated within its own borders.

To accomplish the objective of the 1980 Policy Act, each state had the option of either establishing its own disposal facility or entering into a compact with other states to develop a regional facility. The Act encouraged the formation of compacts and regional solutions to LLW management issues. By January 1, 1986, the deadline for fulfilling the objective of the Act, thirty-six states had formed seven compacts, which were ratified by Congress. These states accounted for approximately 68% of the LLW shipped for disposal in 1986.

No new disposal facilities had been established, however. As it became apparent that no additional states would meet the January 1, 1986, compliance date, the states and compacts began negotiating amendments to the 1980 Act. The result was compromise legislation, the Low-Level Radioactive Waste Policy Amendments Act of 1985, which allows access to the three existing disposal facilities on a limited basis until January 1, 1993. States and regions without their own facilities, are now required to meet a series of milestones. These are deadlines by

which specific actions or tasks must be completed to ensure that new LLW disposal facilities will be developed in states or regions by January 1, 1993. A penalty system of escalating surcharges and eventual loss of access to waste disposal facilities has been devised to foster compliance with the milestones. States and regions that meet the milestones are eligible for a 25% rebate on surcharges. The design of low-level radioactive waste (LLRW) disposal facilities is dependent, among other criteria, on the determination of types, kinds and quantities of waste generated within each compact. This paper discusses the projected types, kinds and quantities of waste for the Southwest and Central States LLRW compacts.

SOUTHWESTERN INTERSTATE COMPACT

The low-level radioactive waste being considered for the Southwestern Compact is based on historical data taken from US Ecology radioactive waste shipment records. All shipments originating from California, Arizona, North Dakota, and South Dakota from 1985 through 1987 have been considered in the analysis. After reviewing shipment records from North and South Dakota, the conclusion was reached that their waste volumes and activities were insignificant and would not be considered further in the analysis.

The various categories of waste (known as waste streams) present in the Southwestern Compact can be divided into two major groups. Table 1 lists those wastes associated with fuel cycle facilities (nuclear power stations) and Table 2 lists those wastes associated with non-fuel cycle facilities. The tables identify those major individual waste streams that, based on historical data, account for more than five percent of the total volume or activity generated for that waste category (fuel cycle or non-fuel cycle), or, based on future projections, will account for more than five percent of the disposed waste volume or activity.

The low-level radioactive wastes (LLRW) expected to be buried at the Southwestern Compact proposed Ward Valley site near Needles, California are produced by several categories of generators including:

1) Nuclear power stations
2) Government
3) Medical
4) Academic Institutions
5) Industry (non-medical)

NUCLEAR POWER STATIONS

The volumes of wastes that can be expected to be produced by the generators in each category are described in the following sections.

Dry solid waste, which consists of contaminated tools, clothing, and equipment, can be grouped into compatible or noncompatible waste. Current operating practices are to attempt to compact as much waste as possible to reduce burial volumes. Thus, several nuclear power stations go through sorting processes to determine which wastes can be compacted to a smaller volume. This type of waste material is usually packaged in 55-gallon drums or some type of large metal box.

Nuclear power stations generate large amounts of radioactive liquids. Typical liquid wastes are from laundry operations, laboratory work, and flushing of various filter media systems. All liquid wastes must be either dewatered, absorbed, or solidified before disposal at the Ward Valley site. Dewatering, absorption, solidification, and packaging of liquid waste, filter media, and resins must be performed by the waste generator in compliance with 10 CFR 61.56, the U.S. NRC Technical Position on Waste Form, and the applicable California regulations, standards, and guides prior to acceptance for disposal at the Ward Valley LLRW disposal facility.

The final type of waste generated at nuclear power stations includes absorbed aqueous and nonaqueous liquids, such as oils, paints, and various other miscellaneous liquids. Any absorbed liquids accepted for disposal at the proposed Ward Valley site will have been absorbed using absorbent approved by the state of California.

GOVERNMENT

Government generators consists of several types of facilities, including naval operations and research facilities. Naval operations generally produce the same types of wastes as nuclear power stations. The research facilities produce various miscellaneous wastes including dry solids, liquids, and others.

MEDICAL

Medical facilities include hospitals, medical research facilities, and any industrial facility that produces products for medical use (such as radiopharmaceutical plants and some biological research labs).

Medical facilities use several radionuclides for diagnostic and therapeutic treatments. These radionuclides are produced artificially in radionuclide production facilities. Normally, LLRW is generated in the form of glass, plastic, trash, and small quantities of metal and liquid. However, it is important to note that these small quantities of liquids may possess very high concentrations of various radionuclides. The proposed Ward Valley site is projected to receive large amounts of tritium (H-3) wastes from several radiopharmaceutical plants. This tritium wastes is usually sealed in structural concrete within stainless steel 2R containers which are placed within 55-gallon drums or within liners.

Other wastes produced by medical facilities include liquids scintillation vials and fluids. The vials, which are made of glass and occasionally polyethylene, are usually filled to less than 50 mL with fluid. Only scintillation fluid that are rendered non-hazardous and which have been packaged in sufficient absorbent material to absorb twice the volume of the liquid or solidified in compliance with 10 CFR 61.56(a)(2) and (8) will be accepted for disposal.

ACADEMIC INSTITUTIONS

Colleges and universities produce various wastes as a result of research projects which can cover a wide variety of fields including medicine, nuclear fuel cycle, and biology.

Probably the largest amount of waste generated by academic institutions is from biological research programs. This waste consists of animal carcases, animal bedding, excreta,

and vegetation and culture media. These types of wastes are usually disposed of after being packed in lime and absorbent material.

Several universities operate neutron activation analysis (NAA) facilities that generate small amounts of waste material in the form of activated metal foils, miscellaneous trash used in lab procedures, and other small amounts of dry solid and liquid wastes. The liquid wastes are packed in absorbents or solidified before disposal.

Another type of waste generated at academic institutions is produced from accelerator target bombardment. Accelerator targets are used to produce radionuclides by direct bombardment with charged practical beams or by indirect reactions of the target fragments with other materials. Spent targets are commonly made of titanium foils containing absorbed tritium.

INDUSTRY (NONMEDICAL)

The last category of waste generators is the industrial group. These facilities manufacture items such a density gauges, well logging devices, radiography sources, X-ray fluorescence tubes, and static eliminators.

The waste generated from these operations include dry solids, liquids, and industrial and institutional sealed sources that have been spent.

WASTE PROJECTIONS

Thirty-year projections have been made of the waste expected to be disposed of at the proposed Ward Valley facility. The basic assumption is that the character and technology of the industry will not change thus the current disposal trends will continue to hold constant for the 30-year operational life of the facility.

Actual radioactive waste shipment record were used as baseline data for the 30-year projection analyses made for the proposed facility. The radioactive waste shipment records from 1985 through 1987 were sorted by month into 36 monthly summaries for four categories:

1. Fuel cycle generator volume
2. Fuel cycle activity
3. Nonfuel cycle generator volume
4. Nonfuel cycle generator activity.

These four categories were plotted to determine if any further modifications were needed on the data before performing the actual 30-year projections. From these plots, it was determined that some of the data needed to be smoothed in order to obtain acceptable results from the projection analyses. The fuel cycle data, for example, showed large variations in monthly volumes shipped to the disposal facilities. Further investigation revealed that the peak months for waste volume shipped immediately preceded waste burial regulatory changes or price increases at the disposal facilities. Therefore, the waste generators were apparently attempting to deplete their waste inventory while the burial and price conditions were more favorable to them. This had the opposite effect on the months immediately following the regulatory or price changes. These months showed large reductions in the amounts of waste shipped for burial.

Standard mathematical models normally do not handle large fluctuations like these satisfactorily. Thus, the data was smoothed in order to provide better baseline data for the 30-year projections.

The smoothing technique was a simple five term averaging model. The resultant data maintained the general trends of the original data but without the large fluctuations caused by regulatory changes or price increases. Projections were performed using standard S-curve models. The S-curve model is based on the function $Z = \exp(a + b/t)$ and fits an S-shaped curve through the data.

The coefficients (a and b) were obtained by least squares after taking the natural logarithm of Z. This trend analysis resulted in 30-year summaries for the four waste categories discussed above. These summaries are shown in Tables 3 and 4, along with the information on decontamination wastes and tritium wastes.

Tritium wastes are a major source of activity projected to be buried at the proposed Ward Valley facility. Large quantities of tritium wastes are generated at several pharmaceutical companies in California. The tritium waste accounts for over 80 percent of the total activity (but less than 6 percent of the volume) projected to be buried at the facility. The projected activities and volumes of the tritium wastes were determined from waste surveys sent to the two major generators of this type of waste.

In addition to analyzing the waste projections based on he different waste generator categories, the projections were also broken down into the different types of waste (or waste streams) expected to be produced.

Tables 5 and 6 depict annual volumes of the various waste streams for fuel cycle and nonfuel cycle generators, projected to be buried at the proposed Ward Valley facility. These volumes are based on total projected volumes (from Table 4) as modified by individual waste stream average volume percentages (from Tables 1 and 2). Again, the assumption has been made that these volume percentages will not change over the operational lifetime of the proposed facility.

Tables 7 and 8 show the annual activities of the various waste streams projected to be buried at the proposed Ward Valley facility. These concentrations are based on overall activities (Table 3) as modified by individual waste stream average activity percentages given in Tables 1 and 2.

The radionuclides used in this analysis are listed in Table 9. The activity shown is in the activity projected to be present in the disposal trenches at the proposed Ward Valley facility at the end of its 30-year operational lifetime (this activity includes radioactive decay and buildup through the 30-year period). The percent column indicates the percentage of the activity that the individual radionuclide accounts for in relation to the total activity. The tritium (H-3) waste accounts for a large percentage of the total activity. Because of this high total activity, the H-3 wastes are treated as a separate waste stream in the analysis.

All radionuclides shipped for burial during the reference period were totaled according to the various fuel cycle and nonfuel cycle waste streams. An assumption was made that these radionuclides would continue to be produced during the 30-year operational life of the proposed

Ward Valley disposal facility. This allowed a projection to be made on the concentrations of radionuclides in each waste stream. These concentrations are shown in Tables 10 and 11.

Note that the values in Tables 10 and 11 include only the radionuclides that were indicated on the actual radioactive waste shipment manifests from the US Ecology database. Thus, no decay of short-lived radionuclides or ingrowth of daughter products is shown unless they were shown on the original manifests.

Class A waste is generally composed of radionuclides with short half-lives. This type of waste poses the smallest threat to the environment since the majority of its activity decays relatively quickly after burial.

Class B waste is composed of a mixture of radionuclides, some with short half-lives and some with longer half-lives. The short-lived radionuclides are usually of very high activities and are placed into this classification as a safety precaution. The longer-lived radionuclides are composed of some transuranics and other radionuclides that are relatively mobile in the environment.

Class C waste is composed of radionuclides that could potentially pose the greatest threat to the environment. This type of waste is primarily composed of transuranic radionuclides and high concentrations of relatively mobile radionuclides.

At the proposed Ward Valley facility, Class B and C wastes are disposed of in a separate trench from the majority of the Class A wastes. In accordance with guidance received from the California Department of Health Services (DHS), stabilized Class A wastes with a contact dose rate greater than 30 R/hr are buried with the B and C wastes in the dedicated "BC30" trench.

The waste streams previously identified can be further broken down to provide more detailed information concerning the wastes to be buried at the proposed Ward Valley facility. Several assumptions must be made to provide a basis for the detailed breakdown. First, historical data is used to provide percentage breakdowns on waste classes, waste streams, and radionuclides. These initial breakdowns are assumed to hold constant for the life of the facility. This is believed to be a valid assumption, as major generator waste production rates are not expected to change significantly during the operational life of the proposed facility.

Thus, the 3-year (from 1985 through 1987) average breakdown of wastes by classification is a good measure of the projected breakdown. These 3-year averages and the resulting projections are shown in Table 12.

FUTURE GENERATION RATE ALTERATIONS

Available information from waste generator survey questionnaires completed by major generators on waste processing techniques indicates that there will be little change over the first 5 years of operation at the Ward Valley facility. This implies that the current volume reduction (VR) and volume increase (VI) factors achieved by various waste processing techniques will remain fairly constant. Volume reduction is achieved at fuel cycle facilities through the use of sorting of dry active wastes (DAW), compacting wastes as appropriate, and evaporation of water from liquid wastes. A volume increase of the waste results from most solidification techniques

(with the exception of bitumen) and packaging systems. Unless otherwise noted, all data presented in this section is based on "as-shipped for burial" (processed) volumes of wastes.

NUCLEAR POWER STATION DECONTAMINATION WASTES

Nuclear power stations may periodically generate an additional type of waste *called* "decontamination waste." Predominately resins containing chelating agents, their kind of waste is discussed below.

During the operation of a nuclear power station, a very thin metal oxide layer forms in several systems of the plant, including the reactor, steam, and water piping systems. This layer, often referred to as "crud," is predominantly an iron-nickel oxide which contains radioactivity mostly in the form of Cobalt-58 and Cobalt-60. These radionuclides are significant gamma radiation emitters and can severely limit the normal maintenance and plant inspection procedures necessary for the safe operation of the facility.

Several chemical cleaning agents have been developed that greatly increase the ability to remove this layer of crud. These chemical agents form complex ions containing the radioactive metal ions from the crud. However, the resulting complex ions are highly soluble and must be handled in a different manner during subsequent burial. Being soluble, the chelated waste may exhibit enhanced transport and mobility properties in the soil, thus potentially migrating faster than other nonchelated wastes buried in the same area. Furthermore, these complex ions have greater bonding potential than the standard waste forms that may be buried at the proposed facility. Since the bonding properties of the chelating agent are stronger than the surrounding backfill material, any radionuclides that may have leaked from other nearby wastes may become attached to the chelate's complex ion structure. These radionuclides would then migrate at the same increased rate as the original chelated radionuclides. The Ward Valley facility's operating procedures shall require the separation of wastes containing chelating agents from other waste in the disposal trenches; the distance between chelate and nonchelate waste is expected to be a minimum of 10 feet. This practice reduces the potential for nonchelated radionuclides becoming attached to the chelated radionuclides.

Little information is available concerning the amount of decontamination wastes containing chelating agents to be disposed of at the facility. Therefore, generic data from NUREG-0782 (NRC81) has been used to project the chelated waste volumes. NUREG-0782 assumes that a decontamination of the primary coolant system at a nuclear power station occurs every 5 to 10 years. Conservatively assuming that a full decontamination is performed every 7 years (an average of the 5 and 10 year estimates) results in the generation of approximately 68,700 cubic feet (processed volume of chelated wastes) to be disposed of over the lifetime of the Ward Valley facility.

This volume is obtained in the following manner. NUREG-0782 estimates that 1677 cubic feet of resin waste are generated as a result of decontaminating the primary coolant system of a typical nuclear reactor. There are nine nuclear reactors in the Southwestern Compact. Conservatively, an assumption is made that all nine reactors perform their first decontamination in 1991, thus resulting in a total of 41 decontaminations over the lifetime of the proposed Ward Valley facility. This number of decontaminations takes into account the projected shutdowns of Rancho Seco Nuclear Generating Station and San Onofre One Nuclear Station. It also takes into

account the expected decontaminations that these two reactors will perform during the initial phases of shutdown. This waste is listed as DECON in Tables 4 and 5. Current trends in the nuclear industry are to use a decontamination process developed in Britain known as low oxidation metal ions (LOMI). This process is based on picolinic acid and vanaddium, a dissolving agent classified as a chelate.

NUCLEAR POWER STATION DECOMMISSIONING WASTES

Another significant source of waste is generated from the decommissioning of nuclear power stations. Six nuclear power stations are projected for shutdown in the 30-year lifetime of the proposed Ward Valley facility. These plants are the 63 megawatt electric (MWe) Humboldt Bay Reactor (currently shut down and in a safe storage mode), the 916 MWe Rancho Seco Nuclear Station, the 436 MWe San Onofre One Nuclear Station, the 1084 MWe Diablo Canyon Unit One Nuclear Station, the 1106 MWe Diablo Canyon Unit Two Nuclear Station, and the 58.5 MWe Pathfinder reactor located in South Dakota (currently in a safe storage mode). The analysis performed assumes that these nuclear stations go into a safe storage mode following shut down and primary coolant system decontamination. Thus, no additional waste volumes or activities are included for nuclear station decommissioning activities.

No inventory limitations are currently proposed for carbon-14, tritium, technetium-99, iodine-129, or the transuranic species. The waste projections result in total site inventories that do not adversely affect the environment; i.e., dose rates resulting from buried wastes are expected to be significantly less than the performance objectives of 10 CFR 61.41. Therefore, there is no basis for placing limitations on the amounts of any radionuclides.

WASTE DURING CLOSURE PERIOD

The proposed Ward Valley facility is expected to have a nominal 5-year closure period immediately following the operational period. During the first 12 to 18 months of this closure period, the site cleanup and building and structure removal is performed. The generation of any radioactive wastes (due to ground or building surface contamination) is expected to occur during the first six months of this period. However, because of the procedures utilized to unload and dispose of radioactive wastes during facility operation, there is no significant waste projected to be generated during the closure of the facility. Thus, a minimal volume of radioactive waste, less than 75 cubic feet, is projected to be generated during closure activities at the site.

DISPOSAL CAPACITY

Based upon regulatory guidance issued by the California Department of Health Services (DHS) there are dedicated trenches for Class A waste. Class B and C wastes, which have longer periods of concern, and must meet additional waste form and stability requirements are disposed of separately from the Class A waste that does not meet these stability requirements. Additionally, the DHS has determined that all wastes with a contact dose rate of 30 R/hour or more, irrespective of class, must meet the regulatory stability requirements and be disposed of in the trench with the Class B and C wastes. Four Class A trenches and one trench for the Class B and C waste, the BC30 trench, are planned.

To ensure that all waste be buried beneath the deepest projected scouring action caused by a Probable Maximum Flood (PMF) all wastes are buried at least 20 feet below the original ground surface at the time of completion. Trench dimensions at the original ground surface depend on the depth and side slope. Soil stability may allow side slopes of one horizontal to one vertical. This is the case presumed for the optimistic calculation of the potential disposal capacity. The trench design may require a 1.5 horizontal to 1.0 vertical side wall design. This is the case presumed for the pessimistic calculation of the potential disposal capacity. Another parameter that effects the potential disposal volume is the efficiency in the use of trench volume. The efficiency is dependent on the method of disposal, waste package design, and the amount of backfill required to shield waste packages. At the Ward Valley facility the waste packages will be stacked using front-end loaders. Experience at the US Ecology LLRW disposal facility in Beatty, Nevada has shown that efficiences range from about 25 to about 33 percent. The four Class A trenches are approximately 1,546 feet long by 290 feet wide at grade by 60 feet deep. An optimistic calculation results in a potential volume of 2.4 E+7 cubic feet for the four Class A trenches. A pessimistic calculation results in a potential disposal volume of 6.8 E+6 cubic feet for the four Class A trenches. The single BC30 trench is 1,546 long by 226 feet wide at the surface and 42 feet deep. A shallower excavation is used to ensure that any moisture accumulation in a Class A trench does not migrate into the Class BC30 trench. The optimistic calculation of the BC30 trench disposal capacity results in 2.5 E+6 cubic feet while pessimistic calculation results in a capacity of 8.5 E+5 cubic feet.

CENTRAL MIDWEST INTERSTATE LLRW COMPACT

The majority of the low-level radioactive waste (LLRW) generated within the Central Midwest Compact region (the states of Illinois and Kentucky) is generated by the nuclear power reactors within the State of Illinois. Wastes from these reactors constitute approximately 80 percent of the volume, and more than 99 percent of the radioactivity in the LLRW from the Central Midwest region. It is projected that approximately 185,000 cubic feet per year of LLRW from power reactors will be sent to the Compact's disposal facility during the early years of its operation, beginning about 1993. During the 21st century the 14 reactors in the state of Illinois should begin to be shut down. As this process continues during the subsequent 30 years, the volumes of waste sent to the Compact's disposal facility could increase significantly. The increase will be dependent on the methods used for decontamination and dismantlement.

The LLRW generated in the Central Midwest region is currently treated by the waste generators to achieve a significant reduction in volume. It is estimated that if current treatment practices continue at the time the regional disposal facility begins operation, approximately 480,000 cubic feet of waste that will be generated annually will be reduced to under 200,000 cubic feet, a reduction by a factor of 2.4 (RAE87). The only way to significantly reduce the activity in the LLRW is to delay the decontamination and dismantlement of the power reactors for a period of decades. Figure 1 illustrates the projected history of volumes of waste entering the regional disposal facility for two scenarios for reactor dismantlement. The first scenario shows the volumes if the reactors are dismantled immediately after the shutdown and the second scenario where the reactors are dismantled after 50 years of storage.

CHARACTERISTIC OF LLRW IN THE CENTRAL MIDWEST COMPACT

A data base was assembled containing the projected characteristics of LLRW going to the disposal facility in 1993. The two primary sources of information in the data base were the 1986 annual surveys of waste generators conducted by the Illinois Department of Nuclear Science (IDNS) and the Kentucky Division of Radiation and Product Safety. The IDNS survey requested projections of volumes and activities of waste that will be produced through 1993. This information was used to forecast waste from Illinois generators. The 1986 volumes and activities of waste generated in the state of Kentucky were used as the projection for the early years of the disposal facility operation (RAE87).

The Illinois and Kentucky surveys were supplement and expanded by a number of telephone calls to the larger waste generators or to any waste generators who had a significant inconsistency in the information reported on the survey forms. Cognizant officials from each state were also contacted by telephone and asked to identify potential sources of radioactive waste that may have been omitted from the survey forms. Projections of reactor decommissioning wastes, which could be a significant contributor to the wastes in the first half of the twenty-first century, were made using information developed by the U.S. Nuclear Regulatory Commission.

A total of 60 waste generators that account for more than 99 percent by volume of 1986 waste shipments from the Central Midwest region were identified, along with their projected rates of generation and shipment of LLRW for disposal in the early 1990s. A source of LLRW not considered in the studies by IDNS and Kentucky in estimating the volumes and activities for disposal are future generators not presently disposing of LLRW. A significant source of LLRW in the future could be clean-up and decontamination projects of the U.S. DOE and the U.S. EPA. No formal study has been promulgated to study the future generation of LLRW due to actions initiated by these federal agencies. Informal discussions with these agencies has indicated that projections of LLRW generated in the Central Midwest Compact would be sporadic and would most likely be less than 10,000 cubic feet per year or a small fraction of the total waste volumes. The activity generated from these projects is estimated to be very low, a maximum of 50 curies.

POTENTIAL IMPACTS OF VOLUME REDUCTION

The potential for reducing volumes of waste entering the regions disposal facility by supercompaction and incineration was investigated (CMI87). Two scenarios were investigated for both supercompaction and incineration to determine their effect on waste volumes. In the first scenario only supercompactable or incineratable waste which are not currently being treated by compaction or incineration by the waste generator would be sent to a regional treatment facility. In the second scenario, all supercompactable or incineratable waste would be sent to the regional treatment facility even though in some cases the waste is currently being compacted or incinerated by the waste generator. Tables 13 and 14 show the potential impacts on annual waste volumed of supercompaction and incineration at a regional treatment facility. Figure 2 summarizes the analyses of the use of regional treatment facilities for supercompaction and incineration of reactor, nonreactor, and total wastes. It shows the reduction in waste volumes that are achieved for reactor wastes and nonreactor wastes when regional incineration and supercompaction facilities are used. It can be seen that the largest reduction in waste volumes

from the operation of a regional treatment facility come from the treatment of reactor wastes, if those wastes are sent to the treatment facility (RAE87).

DISPOSAL CAPACITY

The options for disposal of LLRW at the Central Midwest Compact and choice of specific design parameters will depend on the characteristics of the disposal site. The disposal site has yet to be chosen by the Compact. The disposal site, the allowable waste forms, and disposal unit design have yet to be decided by the Illinois Department of Nuclear Safety and the Compact Commission. These subjects are undergoing extensive scrutiny at this time. Prototype designs that have been reviewed include above and below ground vaults, modular concrete canisters, earth mounded concrete bunkers, augered holes, and mined cavities. Below-ground vaults, modular concrete canisters, earth mounded concrete bunkers, and augered holes are considered to be the most likely to be licensed.

CENTRAL INTERSTATE LLRW COMPACT

The Low-Level Radioactive Waste Amendment Act of 1985 (Public Law 99-240) revised the original act and granted Congressional consent to the states of Arkansas, Kansas, Louisiana, Nebraska and Oklahoma to join together and form the Central Interstate Low-Level Radioactive Waste Compact. In 1987 the Compact Commission selected US Ecology to site, design, license, construct, operate, and close a facility for managing the nonfederal, low-level radioactive and mixed waste generated in the Compact region. US Ecology chose Bechtel National, Inc. (BNI) to be its prime subcontractor. BNI performed a survey of the low-level radioactive waste generators to determine an estimate of the annual LLRW volumes generated in the Compact. The results of the survey are very preliminary and will be augmented by further research on this subject. The results of the survey are listed in Table 15. The survey results were used in preparing initial facility design. This initial design consisted of four above-grade vaults for Class B and C waste with a capacity of 100,000 cubic feet each, one vault for Class B and C waste with a capacity in the range of 65,000 to 75,000 cubic feet, and one small mixed-waste vault. Review of these plans by the waste generators resulted in the requests for the mixed-waste disposal capacity to be increased to 50,000 cubic feet during the 30 year operating period of the disposal facility. The capacity of the aboveground disposal vaults and the waste generation rate data on which the capacity requirements are based is still in a dynamic state and may be modified during the licensing period.

REFERENCES

[1] NRC81 United States Nuclear Regulatory Commission, Draft Environmental Impact Statement on 10 CFR Part 61, *Licensing Requirements for Land Disposal of Radioactive Waste,* NUREG-0782, National Tech. Information Service, Springfield, Virginia, 22161.

[2] RAE87 Central Midwest Interstate Low-Level Radioactive Waste Commission, *Potential Impacts of Source and Volume Reduction Techniques on the Central Midwest Compact's Waste Management System,* prepared by Rogers and Associates Engineering Corp., October, 1987.

TABLE 1

FUEL CYCLE WASTE STREAMS

WASTE DESCRIPTION	MAJOR WASTE STREAM	PERCENTAGE OF TOTAL VOLUME	PERCENTAGE OF TOTAL ACTIVITY
dewatered resin	YES	7.0	44.2
solidified liquids	YES	17.9	1.7
filter media (sludge/cartridges)	NO	1.8	3.8
dry active solid wastes (DAW)	YES	64.9	25.4
solidified resin	YES	8.4	24.9
decontamination wastes	NO	see note 1	see note 1

NOTE 1: The primary reactor coolant decontamination waste stream has been intentionally omitted from this table since the available data concerning this waste stream is limited. Actual projections concerning this waste stream can be seen in Tables 3 and 4.

TABLE 2

NON-FUEL CYCLE WASTE STREAMS

WASTE DESCRIPTION	MAJOR WASTE STREAM	PERCENTAGE OF TOTAL VOLUME	PERCENTAGE OF TOTAL ACTIVITY	MAJOR GENERATORS
solidified liquids	YES	17.5	2.3	radiopharmaceutical companies, labs
biological wastes	NO	3.5	<0.1	research centers and labs
dry active solid wastes	YES	64.7	88.5	all non-fuel cycle generators
tritium wastes	YES	see note 1	see note 1	radiopharmaceutical companies
ion exchange resin	YES	1.0	6.6	government and academic institutes
absorbed aqueous liquids	YES	8.2	2.5	research centers and labs
absorbed non-queous liquids	NO	1.5	<0.1	all non-fuel cycle generators
non-aqueous liquids in vials	NO	3.6	<0.1	research centers and labs

NOTE 1: The tritium waste activity has been intentionally omitted from this table since it would drastically skew the data. The amount of tritiated waste projected to be buried at the facility accounts for more than eighty percent of the total projected burial activity.

TABLE 3

YEARLY ACTIVITY PROJECTIONS
FOR VARIOUS WASTE CATEGORIES/STREAMS

****All activities are given in curies****

YEAR	FUEL CYCLE	NON-FUEL CYCLE	DECON	TRITIUM	TOTALS
1991	557.6	935.8	12870.0	30400.0	44763.4
1992	555.4	936.9	12870.0	33840.0	48202.3
1993	553.8	937.7	12870.0	37544.0	51905.5
1994	552.5	938.3	12870.0	41538.4	55899.2
1995	551.4	938.9	12870.0	45852.2	60212.5
1996	550.6	939.3	12870.0	50517.4	64877.3
1997	549.8	939.6	12870.0	55569.1	69928.5
1998	549.2	940.0	12870.0	61046.0	75405.2
1999	548.7	940.2	12870.0	66990.6	81349.5
2000	548.2	940.4	12870.0	73449.7	87808.3
2001	547.8	940.7	12870.0	80474.7	94833.2
2002	547.5	940.8	12870.0	88122.2	102480.5
2003	547.1	941.0	12870.0	96454.4	110812.5
2004	546.8	941.2	12870.0	105539.8	119897.8
2005	546.6	941.4	12870.0	115453.8	129811.8
2006	546.4	941.5	12870.0	126279.2	140637.1
2007	546.1	941.6	12870.0	138107.1	152464.8
2008	493.5	941.7	6545.0	151037.8	159018.0
2009	469.7	941.8	6545.0	165181.6	173138.1
2010	469.6	941.9	6545.0	180659.8	188616.3
2011	469.4	941.9	6545.0	197605.8	205562.1
2012	469.3	942.0	6545.0	216166.4	224122.7
2013	469.1	942.1	6545.0	236503.0	244459.2
2014	469.0	942.1	6545.0	258793.3	266749.4
2015	468.9	942.2	6545.0	283232.6	291188.7
2016	468.9	942.2	6545.0	310035.9	317992.0
2017	468.8	942.3	6545.0	339439.5	347395.6
2018	468.7	942.3	6545.0	371703.5	379659.5
2019	468.6	942.4	6545.0	407113.9	415069.9
2020	468.5	942.4	6545.0	445985.3	453941.2
TOTALS	15467.5	28222.6	303875.0	4810637.0	5128202.1

TABLE 4

YEARLY VOLUME PROJECTIONS
FOR VARIOUS WASTE CATEGORIES/STREAMS

****All volumes are given in cubic feet****

YEAR	FUEL CYCLE	NON-FUEL CYCLE	DECON	TRITIUM	TOTALS
1991	50693.4	82511.1	2436.9	1437.5	137078.9
1992	50659.3	82439.9	2436.9	1585.0	137121.1
1993	50633.3	82385.4	2436.9	1746.5	137202.1
1994	50612.6	82342.4	2436.9	1923.4	137315.3
1995	50595.9	82307.6	2436.9	2117.2	137457.6
1996	50582.2	82278.9	2436.9	2329.7	137627.7
1997	50570.7	82254.8	2436.9	2562.7	137825.1
1998	50560.8	82234.2	2436.9	2818.2	138050.1
1999	50552.4	82216.5	2436.9	3098.5	138304.3
2000	50545.0	82201.0	2436.9	3406.1	138589.0
2001	50538.4	82187.5	2436.9	3743.7	138906.5
2002	50532.7	82175.4	2436.9	4114.3	139259.3
2003	50527.6	82164.7	2436.9	4521.2	139650.4
2004	50522.9	82155.1	2436.9	4968.1	140083.0
2005	50518.7	82146.4	2436.9	5458.9	140560.9
2006	50514.9	82138.5	2436.9	5998.0	141088.3
2007	50511.7	82131.4	2436.9	6590.3	141670.3
2008	45663.3	82124.8	2102.3	7241.1	137131.5
2009	43462.0	82118.7	2102.3	7956.2	135639.2
2010	43459.7	82113.2	2102.3	8742.1	136417.3
2011	43457.7	82108.1	2102.3	9605.8	137273.9
2012	43455.8	82103.3	2102.3	10555.1	138216.5
2013	43453.8	82098.9	2102.3	11598.6	139253.6
2014	43452.4	82094.8	2102.3	12745.7	140395.2
2015	43450.4	82090.9	2102.3	14006.8	141650.4
2016	43449.1	82087.3	2102.3	15393.2	143031.9
2017	43447.6	82083.9	2102.3	16917.5	144551.3
2018	43446.4	82080.7	2102.3	18593.5	146222.9
2019	43445.1	82077.7	2102.3	20436.4	148061.5
2020	43444.0	82074.8	2102.3	22462.8	150083.9
TOTALS	1426759.8	2465527.9	68757.2	234674.1	4195719.0

TABLE 5

FUEL CYCLE GENERATOR
ANNUAL WASTE STREAM VOLUMES

** all volumes are in cubic feet **

YEAR	DRY ACTIVE WASTES (DAW)	SOLIDIFIED LIQUIDS	FILTER MEDIA	DEWATERED RESINS	SOLIDIFIED RESINS	TOTALS
1991	32896.2	9080.2	934.0	3528.3	4254.7	50693.4
1992	32874.1	9074.1	933.3	3525.9	4251.9	50659.3
1993	32857.2	9069.4	932.9	3524.1	4249.7	50633.3
1994	32843.8	9065.7	932.5	3522.7	4247.9	50612.6
1995	32833.0	9062.7	932.2	3521.5	4246.5	50595.9
1996	32824.0	9060.3	931.9	3520.6	4245.4	50582.2
1997	32816.6	9058.2	931.7	3519.8	4244.4	50570.7
1998	32810.2	9056.4	931.5	3519.1	4243.6	50560.8
1999	32804.7	9054.9	931.4	3518.5	4242.9	50552.4
2000	32799.9	9053.6	931.2	3518.0	4242.3	50545.0
2001	32795.7	9052.4	931.1	3517.5	4241.7	50538.4
2002	32792.0	9051.4	931.0	3517.1	4241.2	50532.7
2003	32788.6	9050.5	930.9	3516.8	4240.8	50527.6
2004	32785.6	9049.7	930.8	3516.4	4240.4	50522.9
2005	32782.9	9048.9	930.7	3516.1	4240.1	50518.7
2006	32780.4	9048.2	930.7	3515.9	4239.7	50514.9
2007	32778.3	9047.6	930.6	3515.7	4239.5	50511.7
2008	29632.1	8179.2	841.3	3178.2	3832.5	45663.3
2009	28203.6	7784.9	800.7	3025.0	3647.8	43462.0
2010	28202.1	7784.5	800.7	3024.8	3647.6	43459.7
2011	28200.8	7784.1	800.7	3024.7	3647.4	43457.7
2012	28199.5	7783.8	800.6	3024.6	3647.3	43455.8
2013	28198.3	7783.4	800.6	3024.4	3647.1	43453.8
2014	28197.3	7783.2	800.6	3024.3	3647.0	43452.4
2015	28196.1	7782.8	800.5	3024.2	3646.8	43450.4
2016	28195.2	7782.6	800.5	3024.1	3646.7	43449.1
2017	28194.2	7782.3	800.5	3024.0	3646.6	43447.6
2018	28193.5	7782.1	800.4	3023.9	3646.5	43446.4
2019	28192.6	7781.9	800.4	3023.8	3646.4	43445.1
2020	28191.9	7781.7	800.4	3023.7	3646.3	43444.0
TOTAL	925860.4	255560.7	26286.3	99303.7	119748.7	1426759.8

TABLE 6

NON-FUEL CYCLE GENERATOR
ANNUAL WASTE STREAM VOLUMES

** all volumes are in cubic feet **

YEAR	DRY ACTIVE WASTES (DAW)	SOLIDIFIED LIQUIDS	SOLIDIFIED RESINS	ABSORBED AQUEOUS LIQUIDS	ABSORBED NON-AQUEOUS LIQUIDS	NON-AQUEOUS LIQUIDS IN VIALS	ANIMAL CARCASSES IN LIME	TOTALS
1991	53384.7	14439.4	825.1	6765.9	1237.6	2970.4	2887.8	82511.1
1992	53338.6	14426.9	824.3	6760.0	1236.5	2967.8	2885.3	82439.9
1993	53303.4	14417.4	823.8	6755.6	1235.7	2965.8	2883.4	82385.4
1994	53275.6	14409.9	823.4	6752.0	1235.1	2964.3	2881.9	82342.4
1995	53253.1	14403.8	823.0	6749.2	1234.6	2963.0	2880.7	82307.6
1996	53234.4	14398.8	822.7	6746.8	1234.1	2962.0	2879.7	82278.9
1997	53218.9	14394.5	822.5	6744.8	1233.8	2961.1	2878.9	82254.8
1998	53205.5	14390.9	822.3	6743.2	1233.5	2960.4	2878.1	82234.2
1999	53194.1	14387.8	822.1	6741.7	1233.2	2959.7	2877.5	82216.5
2000	53184.1	14385.1	822.0	6740.4	1233.0	2959.2	2877.0	82201.0
2001	53175.3	14382.8	821.8	6739.3	1232.8	2958.7	2876.5	82187.5
2002	53167.5	14380.7	821.7	6738.3	1232.6	2958.3	2876.1	82175.4
2003	53160.6	14378.8	821.6	6737.5	1232.4	2957.9	2875.7	82164.7
2004	53154.3	14377.1	821.5	6736.7	1232.3	2957.5	2875.4	82155.1
2005	53148.8	14375.6	821.4	6736.0	1232.1	2957.2	2875.1	82146.4
2006	53143.7	14374.2	821.3	6735.3	1232.0	2956.9	2874.8	82138.5
2007	53139.0	14372.9	821.3	6734.7	1231.9	2956.7	2874.5	82131.4
2008	53134.7	14371.8	821.2	6734.2	1231.8	2956.4	2874.3	82124.8
2009	53130.9	14370.7	821.1	6733.7	1231.7	2956.2	2874.1	82118.7
2010	53127.2	14369.8	821.1	6733.2	1231.6	2956.0	2873.9	82113.2
2011	53123.9	14368.9	821.0	6732.8	1231.6	2955.8	2873.7	82108.1
2012	53120.8	14368.0	821.0	6732.4	1231.5	2955.7	2873.6	82103.3
2013	53118.0	14367.3	820.9	6732.1	1231.4	2955.5	2873.4	82098.9
2014	53115.3	14366.5	820.9	6731.7	1231.4	2955.4	2873.3	82094.8
2015	53112.8	14365.9	820.9	6731.4	1231.3	2955.2	2873.1	82090.9
2016	53110.5	14365.2	820.8	6731.1	1231.3	2955.1	2873.0	82087.3
2017	53108.3	14364.6	820.8	6730.8	1231.2	2955.0	2872.9	82083.9
2018	53106.2	14364.1	820.8	6730.6	1231.2	2954.9	2872.8	82080.7
2019	53104.3	14363.6	820.7	6730.3	1231.1	2954.7	2872.7	82077.7
2020	53102.5	14363.1	820.7	6730.1	1231.1	2954.6	2872.6	82074.8
TOTAL	1595197.0	431466.1	24653.7	202171.8	36981.4	88757.4	86291.8	2465527.9

TABLE 7

FUEL CYCLE GENERATOR
ANNUAL WASTE STREAM ACTIVITIES

** all volumes are in cubic millicuries **

YEAR	DRY ACTIVE WASTES (DAW)	SOLIDIFIED LIQUIDS	FILTER MEDIA	DEWATERED RESINS	SOLIDIFIED RESINS	TOTALS
1991	141852.4	9497.8	20951.0	246216.5	139114.8	557632.5
1992	141296.0	9460.5	20868.8	245250.7	138569.2	555445.2
1993	140873.2	9432.2	20806.4	244516.8	138154.5	553783.1
1994	140540.9	9410.0	20757.3	243940.1	137828.6	552476.9
1995	140272.9	9392.0	20717.7	243474.9	137565.8	551423.3
1996	140052.2	9377.3	20685.1	243091.8	137349.4	550555.8
1997	139867.3	9364.9	20657.8	242770.8	137168.0	549828.8
1998	139710.0	9354.4	20634.6	242497.9	137013.8	549210.7
1999	139574.7	9345.3	20614.6	242263.1	136881.1	548678.8
2000	139457.0	9337.4	20597.2	242058.8	136765.7	548216.1
2001	139353.8	9330.5	20582.0	241879.5	136664.4	547810.2
2002	139262.4	9324.4	20568.5	241720.8	136574.7	547450.8
2003	139180.9	9318.9	20556.5	241579.5	136494.9	547130.7
2004	139108.0	9314.0	20545.7	241452.8	136423.3	546843.8
2005	139042.1	9309.6	20536.0	241338.6	136358.8	546585.1
2006	138982.5	9305.6	20527.1	241235.0	136300.3	546350.5
2007	138928.1	9302.0	20519.1	241140.7	136247.0	546136.9
2008	125548.5	8406.2	18543.0	217917.4	123125.6	493540.7
2009	119476.3	7999.3	17645.4	207369.0	117165.6	469655.6
2010	119440.1	7996.8	17640.1	207306.3	117130.2	469513.5
2011	119406.7	7994.6	17635.1	207248.3	117097.4	469382.1
2012	119375.8	7992.5	17630.6	207194.6	117067.1	469260.6
2013	119347.0	7990.6	17626.3	207144.6	117038.8	469147.3
2014	119320.2	7988.8	17622.4	207098.1	117012.5	469042.0
2015	119295.1	7987.1	17618.7	207054.6	116987.9	468943.4
2016	119271.7	7985.6	17615.2	207013.9	116965.0	468851.4
2017	119248.7	7984.1	17612.0	206975.7	116943.4	468763.9
2018	119228.0	7982.7	17608.9	206939.8	116923.1	468682.5
2019	119208.6	7981.4	17606.0	206906.0	116904.0	468606.0
2020	119190.2	7980.2	17603.3	206874.1	116886.0	468533.8
TOTAL	3934711.3	263446.7	581132.4	6829470.7	3858720.9	15467482.0

TABLE 8

NON-FUEL CYCLE GENERATOR
ANNUAL WASTE STREAM ACTIVITIES

** all volumes are in cubic millicuries **

YEAR	DRY ACTIVE WASTES (DAW)	SOLIDIFIED LIQUIDS	SOLIDIFIED RESINS	ABSORBED AQUEOUS LIQUIDS	ABSORBED NON-AQUEOUS LIQUIDS	NON-AQUEOUS LIQUIDS IN VIALS	ANIMAL CARCASSES IN LIME	TOTALS
1991	828066.8	21421.7	61491.2	23714.7	5416.0	189.1	470.8	935896.1
1992	828999.5	21445.9	61560.4	23741.4	5422.1	189.3	471.3	936950.2
1993	829713.6	21464.3	61613.5	23761.8	5426.8	189.4	471.7	937757.3
1994	830278.0	21478.9	61655.4	23778.0	5430.5	189.6	472.0	938395.2
1995	830735.3	21490.8	61689.3	23791.1	5433.5	189.7	472.3	938912.1
1996	831113.5	21500.5	61717.4	23801.9	5436.0	189.7	472.5	939339.5
1997	831431.3	21508.8	61741.0	23811.0	5438.0	189.8	472.7	939698.7
1998	831702.2	21515.8	61761.1	23818.8	5439.8	189.9	472.8	940004.9
1999	831936.0	21521.8	61778.5	23825.5	5441.3	189.9	473.0	940269.1
2000	832139.7	21527.1	61793.6	23831.3	5442.7	190.0	473.1	940499.3
2001	832318.7	21531.7	61806.9	23836.4	5443.8	190.0	473.2	940701.6
2002	832477.3	21535.8	61818.7	23841.0	5444.9	190.1	473.3	940880.9
2003	832618.9	21539.5	61829.2	23845.0	5445.8	190.1	473.3	941040.9
2004	832745.9	21542.8	61838.6	23848.7	5446.6	190.1	473.4	941184.5
2005	832860.7	21545.7	61847.2	23852.0	5447.4	190.1	473.5	941314.2
2006	832964.7	21548.4	61854.9	23854.9	5448.1	190.2	473.5	941431.8
2007	833059.6	21550.9	61861.9	23857.7	5448.7	190.2	473.6	941539.0
2008	833146.4	21553.1	61868.4	23860.1	5449.3	190.2	473.6	941637.1
2009	833226.0	21555.2	61874.3	23862.4	5449.8	190.2	473.7	941727.1
2010	833299.5	21557.1	61879.8	23864.5	5450.3	190.2	473.7	941810.2
2011	833367.5	21558.9	61884.8	23866.5	5450.7	190.3	473.8	941887.0
2012	833430.5	21560.5	61889.5	23868.3	5451.1	190.3	473.8	941958.2
2013	833489.1	21562.0	61893.8	23870.0	5451.5	190.3	473.8	942024.4
2014	833543.7	21563.4	61897.9	23871.5	5451.9	190.3	473.9	942086.2
2015	833594.8	21564.7	61901.7	23873.0	5452.2	190.3	473.9	942143.9
2016	833642.6	21566.0	61905.2	23874.4	5452.5	190.3	473.9	942197.9
2017	833687.5	21567.1	61908.6	23875.6	5452.8	190.3	474.0	942248.7
2018	833729.7	21568.2	61911.7	23876.8	5453.1	190.3	474.0	942296.4
2019	833769.5	21569.3	61914.7	23878.0	5453.3	190.4	474.0	942341.4
2020	833807.1	21570.2	61917.4	23879.1	5453.6	190.4	474.0	942383.8
TOTAL	24970895.6	645986.1	1854306.6	715131.4	163324.1	5701.0	14196.1	28222557.6

TABLE 9

MAJOR NUCLIDES CONSIDERED FOR ANALYSIS

NUCLIDE	ACTIVITY*	PERCENT
AM-241	1.660E+0	<0.1
C-14	2.746E+2	<0.1
CM-243	2.480E-2	<0.1
CM-244	1.433E-1	<0.1
CO-60	1.831E+4	0.4
CS-137	9.702E+3	0.2
FE-55	3.224E+3	<0.1
H -3	4.811E+6	99.3
I -129	9.888E+0	<0.1
NB-94	1.443E-1	<0.1
NI-59	1.793E+1	<0.1
NI-63	1.463E+3	<0.1
NP-237	1.900E-3	<0.1
PU-238	8.317E-1	<0.1
PU-239	4.467E-1	<0.1
PU-240	4.207E-1	<0.1
PU-241	5.098E+1	<0.1
RA-226	1.610E+1	<0.1
RN-222	1.610E+1	<0.1
SR-90	2.245E+3	<0.1
TC-99	2.768E+0	<0.1
TH-231	2.757E+0	<0.1
TH-234	1.726E+2	<0.1
U -235	2.757E+0	<0.1
U -238	1.726E+2	<0.1
TOTALS	4.844E+6	99.9

* All activities are given in curies.

TABLE 10

FUEL CYCLE GENERATOR
RADIONUCLIDE CONCENTRATION BREADKOWN BY WASTE STREAM

** all concentrations are in microcuries per cubic centimeter **

RADIO-NUCLIDE	DRY ACTIVE WASTES (DAW)	SOLIDIFIED LIQUIDS	FILTER MEDIA	DEWATERED RESINS	SOLIDIFIED RESINS	TOTALS
AG-110M	1.049E-02	3.828E-05	1.673E-05	0.000E+00	5.680E-04	1.111E-02
AM-241	0.000E+00	4.283E-09	0.000E+00	1.316E-08	7.767E-07	7.941E-07
BA-140	0.000E+00	5.112E-09	1.127E-05	0.000E+00	0.000E+00	1.128E-05
BE-7	1.484E-06	4.725E-08	5.649E-06	1.316E-08	2.131E-03	2.138E-03
BI-207	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
C-14	4.526E-04	1.313E-05	2.034E-03	1.897E-04	1.754E-03	4.443E-03
CD-109	0.000E+00	0.000E+00	2.552E-08	0.000E+00	0.000E+00	2.552E-08
CE-141	8.904E-06	8.290E-10	1.782E-04	1.841E-05	0.000E+00	2.056E-04
CE-144	7.123E-05	3.454E-09	6.918E-04	1.025E-03	1.721E-05	1.805E-03
CM-242	1.484E-06	3.689E-08	5.427E-07	3.307E-08	1.121E-06	3.217E-06
CM-243	0.000E+00	4.283E-09	0.000E+00	1.316E-08	1.769E-09	1.921E-08
CM-244	0.000E+00	4.283E-09	0.000E+00	1.316E-08	6.537E-07	6.712E-07
CO-57	8.904E-06	1.255E-05	5.243E-04	5.949E-04	2.149E-04	1.355E-03
CO-58	3.300E-02	1.359E-03	2.120E-01	3.721E-01	6.784E-02	6.863E-01
CO-60	1.468E-02	9.860E-04	9.395E-02	2.085E-01	9.206E-02	4.101E-01
CR-51	7.806E-03	0.000E+00	1.820E-01	9.172E-06	9.489E-04	1.908E-01
CS-134	4.541E-04	1.514E-03	1.847E-03	3.422E-01	1.100E-01	4.560E-01
CS-136	0.000E+00	0.000E+00	6.023E-06	8.303E-07	0.000E+00	6.854E-06
CS-137	1.732E-03	4.184E-03	7.273E-03	7.966E-01	4.028E-01	1.213E+00
EU-154	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU-155	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
FE-55	2.667E-02	1.327E-03	3.841E-02	1.071E-01	1.474E-01	3.210E-01
FE-59	2.003E-04	1.133E-07	1.459E-02	7.019E-03	9.945E-03	3.175E-02
H-3	1.046E-03	1.833E-02	1.410E-03	4.949E-03	1.514E-02	4.088E-02
I-129	4.452E-06	6.356E-08	2.077E-05	6.754E-06	1.683E-06	3.372E-05
I-131	1.677E-04	4.654E-05	7.137E-04	2.675E-05	1.219E-03	2.173E-03
I-133	0.000E+00	1.658E-09	0.000E+00	2.133E-09	0.000E+00	3.791E-09
KR-85	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA-140	0.000E+00	5.748E-08	0.000E+00	1.435E-06	1.142E-04	1.157E-04
MN-54	1.658E-03	1.412E-04	2.714E-02	8.546E-02	3.411E-02	1.485E-01
MO-99	0.000E+00	1.658E-09	0.000E+00	0.000E+00	0.000E+00	1.658E-09
NA-24	0.000E+00	9.354E-08	0.000E+00	7.688E-06	0.000E+00	7.781E-06
NB-94	0.000E+00	0.000E+00	2.552E-08	0.000E+00	0.000E+00	2.552E-08
NB-95	3.491E-03	4.463E-08	3.250E-02	2.750E-04	7.147E-04	3.698E-02
NI-59	0.000E+00	0.000E+00	4.811E-05	6.354E-04	6.478E-05	7.483E-04
NI-63	2.553E-02	9.950E-04	3.307E-02	6.492E-02	9.501E-02	2.195E-01
NI-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NP-237	0.000E+00	0.000E+00	0.000E+00	2.133E-09	0.000E+00	2.133E-09
PB-210	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

TABLE 10 (con't)

RADIO-NUCLIDE	DRY ACTIVE WASTES (DAW)	SOLIDIFIED LIQUIDS	FILTER MEDIA	DEWATERED RESINS	SOLIDIFIED RESINS	TOTALS
PU-238	1.484E-06	3.261E-08	2.552E-08	1.316E-08	2.278E-06	3.834E-06
PU-239	0.000E+00	2.059E-08	5.776E-08	1.316E-08	1.967E-06	2.059E-06
PU-240	0.000E+00	4.283E-09	4.970E-08	1.316E-08	1.769E-09	6.891E-08
PU-241	1.172E-04	2.780E-06	2.080E-04	1.794E-04	2.945E-04	8.020E-04
RU-103	0.000E+00	0.000E+00	1.578E-04	0.000E+00	0.000E+00	1.578E-04
RU-106	3.265E-05	0.000E+00	1.268E-05	0.000E+00	0.000E+00	4.533E-05
SB-122	0.000E+00	1.658E-09	0.000E+00	3.092E-06	0.000E+00	3.094E-06
SB-124	1.619E-03	2.297E-03	4.800E-03	1.062E-01	1.558E-04	1.151E-01
SB-125	2.820E-05	1.726E-04	7.208E-04	2.131E-04	8.055E-05	1.215E-03
SN-113	0.000E+00	0.000E+00	1.938E-05	0.000E+00	0.000E+00	1.938E-05
SR-89	0.000E+00	0.000E+00	2.312E-06	1.104E-03	2.826E-06	1.109E-03
SR-90	5.936E-06	1.037E-05	2.554E-05	1.576E-03	1.031E-03	2.649E-03
TA-182	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC-99	8.904E-06	1.064E-07	4.561E-05	8.424E-06	2.209E-06	6.526E-05
TC-99M	1.039E-05	8.290E-10	0.000E+00	2.133E-09	0.000E+00	1.039E-05
TE-125M	1.484E-06	1.072E-05	0.000E+00	0.000E+00	0.000E+00	1.220E-05
XE-131M	0.000E+00	5.112E-09	1.143E-05	0.000E+00	0.000E+00	1.144E-05
XE-133	0.000E+00	1.658E-09	2.934E-05	0.000E+00	0.000E+00	2.934E-05
ZN-65	1.187E-05	8.290E-10	2.367E-03	1.119E-04	1.363E-04	2.627E-03
ZR-95	4.912E-04	2.625E-09	1.918E-02	1.339E-04	3.349E-04	2.014E-02

TABLE 11

NON-FUEL CYCLE GENERATOR
RADIONUCLIDE CONCENTRATION BREADKOWN BY WASTE STREAM

** all concentrations are in microcuries per cubic centimeter **

RADIO-NUCLIDE	DRY ACTIVE WASTES (DAW)	SOLIDIFIED LIQUIDS	SOLIDIFIED RESINS	ABSORBED AQUEOUS LIQUIDS	ABSORBED NON-AQUEOUS LIQUIDS	NON-AQUEOUS LIQUIDS IN VIALS	ANIMAL CARCASSES IN LIME	TOTALS
AC-227	2.101E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.101E-07
AC-228	4.518E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.518E-09
AG-108M	2.600E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.600E-06
AG-110M	1.319E-06	0.000E+00	2.635E-05	4.838E-06	9.548E-08	0.000E+00	0.000E+00	3.261E-05
AL-26	0.000E+00	0.000E+00	0.000E+00	2.712E-05	0.000E+00	0.000E+00	0.000E+00	2.712E-05
AM-241	2.298E-05	3.936E-06	8.157E-04	6.986E-08	0.000E+00	0.000E+00	0.000E+00	8.426E-04
AU-195	1.197E-05	4.092E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.201E-05
BA-133	9.329E-07	8.184E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	9.411E-07
BA-140	0.000E+00	0.000E+00	1.289E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.289E-06
BE-7	1.694E-07	8.184E-09	1.432E-07	1.747E-08	9.548E-08	0.000E+00	0.000E+00	4.338E-07
BI-205	8.358E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	8.358E-07
BI-207	2.666E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.666E-07
BR-82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
C-14	2.138E-03	6.177E-03	8.327E-04	1.277E-02	4.207E-03	4.884E-04	8.405E-04	2.746E-02
CA-45	3.352E-05	2.782E-06	0.000E+00	8.462E-05	0.000E+00	2.029E-06	2.881E-04	4.110E-04
CD-109	3.636E-05	4.062E-05	0.000E+00	3.423E-06	0.000E+00	2.387E-07	1.113E-05	9.178E-05
CE-139	4.518E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.518E-09
CE-141	1.276E-06	3.273E-08	0.000E+00	8.034E-07	0.000E+00	4.336E-06	3.364E-05	4.008E-05
CE-144	4.554E-06	8.559E-05	6.771E-03	5.764E-07	0.000E+00	0.000E+00	0.000E+00	6.862E-03
CF-252	4.518E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.518E-09
CL-36	1.324E-06	4.665E-07	0.000E+00	1.167E-05	4.392E-06	3.819E-06	6.424E-06	2.809E-05
CM-242	2.711E-08	3.273E-08	1.432E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.031E-07
CM-243	5.602E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	5.602E-07
CM-244	2.950E-06	1.473E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.097E-06
CO-57	5.963E-04	3.110E-07	0.000E+00	1.039E-03	8.593E-07	8.828E-05	2.476E-05	1.750E-03
CO-58	1.582E-05	3.368E-05	4.085E-03	2.253E-06	1.685E-04	0.000E+00	0.000E+00	4.306E-03
CO-60	3.726E-01	9.566E-03	1.478E-01	2.876E-03	1.110E-03	3.501E-06	0.000E+00	5.339E-01
CR-51	3.083E-04	2.204E-05	8.031E-04	1.429E-03	1.038E-04	6.970E-05	2.251E-04	2.961E-03
CS-134	3.683E-05	2.561E-04	3.339E-02	1.104E-05	1.719E-05	0.000E+00	0.000E+00	3.371E-02
CS-136	0.000E+00	0.000E+00	2.864E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.864E-07
CS-137	9.416E-02	1.381E-02	1.571E+00	8.696E-04	4.923E-04	1.472E-06	0.000E+00	1.681E+00
CS-141	8.358E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.501E-07	1.286E-06
CU-64	9.036E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	9.036E-09
CU-67	2.101E-07	0.000E+00	0.000E+00	1.799E-06	0.000E+00	0.000E+00	5.189E-05	5.389E-05
EU-152	1.145E-04	1.447E-03	6.144E-02	1.163E-05	1.012E-05	0.000E+00	0.000E+00	6.302E-02
EU-154	1.520E-05	6.211E-05	2.246E-03	9.082E-07	9.548E-08	0.000E+00	0.000E+00	2.324E-03
EU-155	4.491E-05	8.090E-04	3.406E-02	5.921E-06	2.482E-06	0.000E+00	0.000E+00	3.492E-02
FE-55	2.556E-02	1.773E-02	1.340E-01	3.663E-04	9.684E-04	3.978E-07	1.637E-07	1.786E-01

TABLE 11 (con't)

RADIO-NUCLIDE	DRY ACTIVE WASTES (DAW)	SOLIDIFIED LIQUIDS	SOLIDIFIED RESINS	ABSORBED AQUEOUS LIQUIDS	ABSORBED NON-AQUEOUS LIQUIDS	NON-AQUEOUS LIQUIDS IN VIALS	ANIMAL CARCASSES IN LIME	TOTALS
FE-59	1.022E-05	0.000E+00	3.390E-04	3.221E-05	1.442E-05	7.957E-08	6.297E-05	4.589E-04
GA-67	1.543E-05	3.273E-08	0.000E+00	3.423E-06	0.000E+00	4.177E-06	4.321E-05	6.628E-05
GD-153	3.727E-07	0.000E+00	0.000E+00	1.799E-06	0.000E+00	0.000E+00	3.237E-05	3.454E-05
GE-68	2.553E-07	0.000E+00	0.000E+00	9.082E-07	0.000E+00	0.000E+00	0.000E+00	1.163E-06
HF-175	0.000E+00	0.000E+00	6.359E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	6.359E-05
HF-181	0.000E+00	5.729E-08	3.655E-04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.656E-04
HG-203	3.434E-07	8.184E-09	0.000E+00	2.620E-07	0.000E+00	0.000E+00	0.000E+00	6.135E-07
HO-166M	5.568E-06	0.000E+00	0.000E+00	0.000E+00	5.729E-07	0.000E+00	0.000E+00	6.141E-06
I-123	7.073E-06	1.515E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.189E-04	4.411E-04
I-124	8.584E-08	0.000E+00	0.000E+00	0.000E+00	7.066E-06	5.888E-06	1.228E-07	1.316E-05
I-125	6.153E-03	2.630E-04	0.000E+00	1.328E-02	1.774E-04	5.581E-04	2.294E-03	2.273E-02
I-129	7.981E-06	6.203E-06	1.331E-02	3.493E-07	9.548E-08	0.000E+00	0.000E+00	1.332E-02
I-131	1.105E-04	3.026E-05	6.875E-06	1.419E-03	1.719E-05	2.665E-06	1.032E-04	1.689E-03
I-135	6.619E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	6.619E-07
IN-111	2.808E-05	0.000E+00	0.000E+00	2.152E-05	1.766E-05	1.671E-06	3.256E-04	3.946E-04
IN-113M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.339E-05	3.339E-05
IN-114	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	9.002E-07	9.002E-07
IN-114M	3.501E-07	0.000E+00	0.000E+00	8.733E-08	0.000E+00	0.000E+00	1.473E-05	1.517E-05
IR-192	1.204E-06	0.000E+00	0.000E+00	1.939E-06	0.000E+00	0.000E+00	0.000E+00	3.143E-06
K-40	9.962E-07	0.000E+00	0.000E+00	3.877E-06	0.000E+00	0.000E+00	0.000E+00	4.873E-06
K-42	0.000E+00	0.000E+00	0.000E+00	6.462E-06	0.000E+00	3.978E-08	0.000E+00	6.502E-06
KR-85	3.693E-04	3.026E-04	0.000E+00	0.000E+00	7.066E-03	0.000E+00	0.000E+00	7.738E-03
LA-140	0.000E+00	8.184E-09	1.432E-07	1.747E-08	0.000E+00	0.000E+00	0.000E+00	1.689E-07
MN-54	5.944E-05	3.736E-05	2.772E-03	3.930E-06	2.941E-05	0.000E+00	4.910E-07	2.903E-03
MN-56	6.777E-08	9.002E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.578E-07
MO-99	4.706E-05	0.000E+00	0.000E+00	6.462E-06	0.000E+00	1.472E-05	0.000E+00	6.825E-05
NA-22	9.074E-06	1.274E-05	0.000E+00	4.110E-05	9.548E-07	5.172E-07	1.526E-05	7.965E-05
NA-24	1.807E-08	1.555E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.573E-06
NB-94	3.115E-06	0.000E+00	0.000E+00	1.100E-06	0.000E+00	0.000E+00	0.000E+00	4.215E-06
NB-95	9.036E-07	0.000E+00	1.905E-05	8.733E-08	2.492E-05	0.000E+00	3.564E-05	8.060E-05
NI-59	3.482E-04	4.239E-06	1.409E-04	6.637E-07	0.000E+00	0.000E+00	0.000E+00	4.940E-04
NI-63	9.532E-04	9.142E-04	2.170E-02	3.706E-05	1.186E-04	0.000E+00	0.000E+00	2.372E-02
NI-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NP-237	4.292E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.292E-08
NP-239	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
P-32	5.334E-03	2.085E-04	0.000E+00	8.966E-02	6.058E-04	9.530E-04	8.008E-05	9.684E-02
P-33	4.021E-07	0.000E+00	0.000E+00	1.140E-05	0.000E+00	5.172E-07	0.000E+00	1.232E-05
PA-233	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PB-210	7.632E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.580E-07	0.000E+00	7.668E-05
PB-212	4.518E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.518E-09
PM-147	5.571E-04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	5.571E-04
PO-210	3.638E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.228E-07	3.650E-05
PT-195	3.011E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.011E-06
PU-238	1.177E-05	2.030E-06	3.506E-04	2.270E-07	0.000E+00	0.000E+00	0.000E+00	3.646E-04

TABLE 11 (con't)

RADIO-NUCLIDE	DRY ACTIVE WASTES (DAW)	SOLIDIFIED LIQUIDS	SOLIDIFIED RESINS	ABSORBED AQUEOUS LIQUIDS	ABSORBED NON-AQUEOUS LIQUIDS	NON-AQUEOUS LIQUIDS IN VIALS	ANIMAL CARCASSES IN LIME	TOTALS
PU-239	5.390E-06	8.184E-09	2.569E-04	1.747E-08	0.000E+00	0.000E+00	0.000E+00	2.624E-04
PU-240	5.453E-06	0.000E+00	2.569E-04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.624E-04
PU-241	5.533E-04	4.494E-04	2.209E-02	8.925E-06	1.929E-05	0.000E+00	2.537E-06	2.312E-02
RA-226	3.636E-04	0.000E+00	0.000E+00	6.986E-08	0.000E+00	0.000E+00	0.000E+00	3.637E-04
RA-228	8.109E-07	0.000E+00	3.180E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.261E-05
RB-83	1.588E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.588E-06
RB-86	9.241E-06	1.097E-06	0.000E+00	7.021E-06	2.473E-05	2.773E-05	1.228E-07	6.994E-05
RE-184	2.259E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.259E-09
RU-103	1.100E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.683E-05	3.793E-05
RU-105	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	7.775E-07	7.775E-07
RU-106	6.555E-06	0.000E+00	0.000E+00	1.572E-07	0.000E+00	0.000E+00	0.000E+00	6.713E-06
SB-122	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB-124	3.161E-03	6.244E-06	5.299E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.220E-03
SB-125	1.332E-05	0.000E+00	2.721E-06	5.764E-07	9.548E-08	0.000E+00	0.000E+00	1.671E-05
SC-46	1.186E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.071E-05	4.190E-05
SC-47	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE-75	4.235E-06	2.954E-06	0.000E+00	6.991E-05	3.533E-06	0.000E+00	6.670E-06	8.731E-05
SI-32	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN-113	4.427E-06	0.000E+00	5.299E-06	8.209E-07	0.000E+00	4.336E-06	4.292E-05	5.781E-05
SN-119m	9.307E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.211E-05	1.304E-05
SN-126	2.736E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.736E-06
SR-85	1.319E-06	4.910E-08	0.000E+00	1.589E-06	0.000E+00	4.336E-06	1.915E-05	2.644E-05
SR-89	3.840E-08	2.660E-06	0.000E+00	3.249E-06	0.000E+00	0.000E+00	1.363E-05	1.957E-05
SR-90	4.588E-02	1.794E-04	2.982E-01	5.685E-04	6.493E-05	2.387E-07	0.000E+00	3.449E-01
SR-91	4.604E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.604E-06
TA-182	7.387E-07	0.000E+00	2.650E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.724E-05
TB-160	0.000E+00	1.637E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.637E-08
TC-99	4.978E-05	7.365E-07	6.717E-05	1.327E-05	2.397E-05	5.291E-06	1.723E-05	1.774E-04
TC-99M	1.086E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.030E-05	3.305E-04	3.517E-04
TE-123M	2.047E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.047E-06
TH-228	8.652E-07	0.000E+00	0.000E+00	8.733E-08	0.000E+00	0.000E+00	0.000E+00	9.525E-07
TH-230	2.485E-08	0.000E+00	0.000E+00	5.240E-08	0.000E+00	0.000E+00	0.000E+00	7.724E-08
TH-232	3.689E-06	1.620E-06	0.000E+00	1.048E-07	9.548E-08	0.000E+00	0.000E+00	5.509E-06
TL-201	9.562E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.342E-06	1.308E-04	1.437E-04
TL-204	5.376E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	5.376E-07
TL-208	6.777E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	6.777E-09
U-233	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
U-234	1.751E-06	0.000E+00	0.000E+00	1.921E-07	0.000E+00	0.000E+00	0.000E+00	1.943E-06
U-235	2.580E-05	1.141E-04	1.432E-07	1.275E-06	2.026E-04	0.000E+00	0.000E+00	3.440E-04
U-238	3.858E-03	1.221E-04	1.289E-06	3.727E-05	7.333E-05	0.000E+00	8.184E-08	4.092E-03
XE-131M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE-133	5.674E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	9.866E-06	2.113E-04	2.268E-04
Y-88	9.442E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	9.442E-07
Y-90	1.118E-04	1.372E-04	2.982E-01	5.292E-06	2.005E-06	0.000E+00	8.184E-08	2.985E-01

TABLE 11 (con't)

RADIO-NUCLIDE	DRY ACTIVE WASTES (DAW)	SOLIDIFIED LIQUIDS	SOLIDIFIED RESINS	ABSORBED AQUEOUS LIQUIDS	ABSORBED NON-AQUEOUS LIQUIDS	NON-AQUEOUS LIQUIDS IN VIALS	ANIMAL CARCASSES IN LIME	TOTALS
YB-169	9.420E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	9.420E-07
ZN-65	8.483E-05	0.000E+00	0.000E+00	1.231E-04	0.000E+00	1.591E-07	4.583E-06	2.126E-04
ZR-95	8.132E-08	0.000E+00	3.523E-05	1.747E-08	1.404E-05	0.000E+00	0.000E+00	4.937E-05

TABLE 12

PERCENTAGE BREAKDOWN OF WASTE
STREAM VOLUMES BY WASTE CLASSIFICATION

WASTE STREAM	WASTE CLASSIFICATION A UNSTABLE	A STABLE	B	C
FUEL CYCLE				
DAW	94.8	1.9	1.9	1.4
Solidified Liquids	61.4	38.6	0.0	0.0
Filter Media	64.9	20.6	14.5	0.0
Dewatered Resins	67.4	4.8	25.5	2.3
Solidified Resins	63.2	12.7	17.3	6.8
NON-FUEL CYCLE				
DAW	96.5	2.5	0.4	0.6
Solidified liquids	84.9	3.6	9.8	1.7
Solidified Resins	100.0	0.0	0.0	0.0
Absorbed Aqueous Liquids	100.0	0.0	0.0	0.0
Absorbed Non-Aqueous Liquids	100.0	0.0	0.0	0.0
Non-Aqueous Liquids in Vials	100.0	0.0	0.0	0.0
Animals Carcasses in Lime	100.0	0.0	0.0	0.0

TABLE 13

POTENTIAL IMPACTS ON ANNUAL WASTE VOLUME OF INCINERATION AT REGIONAL TREATMENT FACILITIES (Source RAE87)[1]

Scenario	Volume of Waste Treated in The Disposal Facility	Total Volume of Waste for Disposal
Current Treatment Practices Continue for Waste That is Currently Being Treated	27,000 ft^3	154,000 ft^3 (83 percent)[2]
All Appropriate Wastes are Incinerated	249,000 ft^3	136,000 ft^3 (74 percent)

1. The estimated annual volume of LLW in the Central Midwest Compact in the 1990's is 185,000 ft^3 if projected treatment practices are followed. Projected treatment processes are essentially the processes now in use.

2. Numbers shown in parenthesis state the percent of base case annual waste volume (current treatment practices continue).

3. As currently treated, these wastes have a volume of about 74,000 ft^3 when shipped for disposal.

TABLE 14

POTENTIAL IMPACTS ON ANNUAL WASTE VOLUME OF INCINERATION AT REGIONAL TREATMENT FACILITIES (Source RAE87)[1]

Scenario	Volume of Waste Treated in The Disposal Facility	Total Volume of Waste for Disposal
Current Treatment Practices Continue for Waste That is Currently Being Treated	27,000 ft^3	162,000 ft^3 (88 percent)[2]
All Appropriate Wastes are Incinerated	247,000[3] ft^3	159,000 ft^3 (86 percent)

1. The estimated annual volume of LLW in the Central Midwest Compact in the 1990's is 185,000 ft^3 if projected treatment practices are followed. Projected treatment processes are essentially the processes now in use.

2. Numbers shown in parenthesis state the percent of base case annual waste volume (current treatment practices continue).

3. As currently treated, these wastes have a volume of about 67,000 ft^3 when shipped for disposal.

TABLE 15

ESTIMATE OF THE ANNUAL LLRW VOLUMES GENERATED
IN THE CENTRAL INTERSTATE COMPACT

	LLRW Generator	Class A(ft^3/Yr)	Class B (ft^3/Yr)	Class C (ft^3/Yr)	Mixed (ft^3/Yr)
Nuclear Power Plants					
	Cooper	11,343 ft^3/yr	165 ft^3/yr	0 ft^3/yr	30 ft^3/yr
	Arkansas 1 and 2	7,410 ft^3/yr	1,090 ft^3/yr	0 ft^3/yr	Unknown
	Wolf Creek	4,070 ft^3/yr	240 ft^3/yr	0 ft^3/yr	Unknown
	River Bend	11,738 ft^3/yr	0 ft^3/yr	0 ft^3/yr	Unknown
	Waterford 3	7,789 ft^3/yr	360 ft^3/yr	7 ft^3/yr	Unknown
	Fort Calhoun	3,968 ft^3/yr	142 ft^3/yr	0 ft^3/yr	
	Generic Act. Metals			7 ft^3/yr	
Medical Facilies	1,521 ft^3/yr	0	0	113 ft^3/yr	
Industrial Facilities	3,045 ft^3/yr	0	0	0	
TOTALS	50,875 ft^3/yr	1,997 ft^3/yr	7 ft^3/yr	143 ft^3/yr	

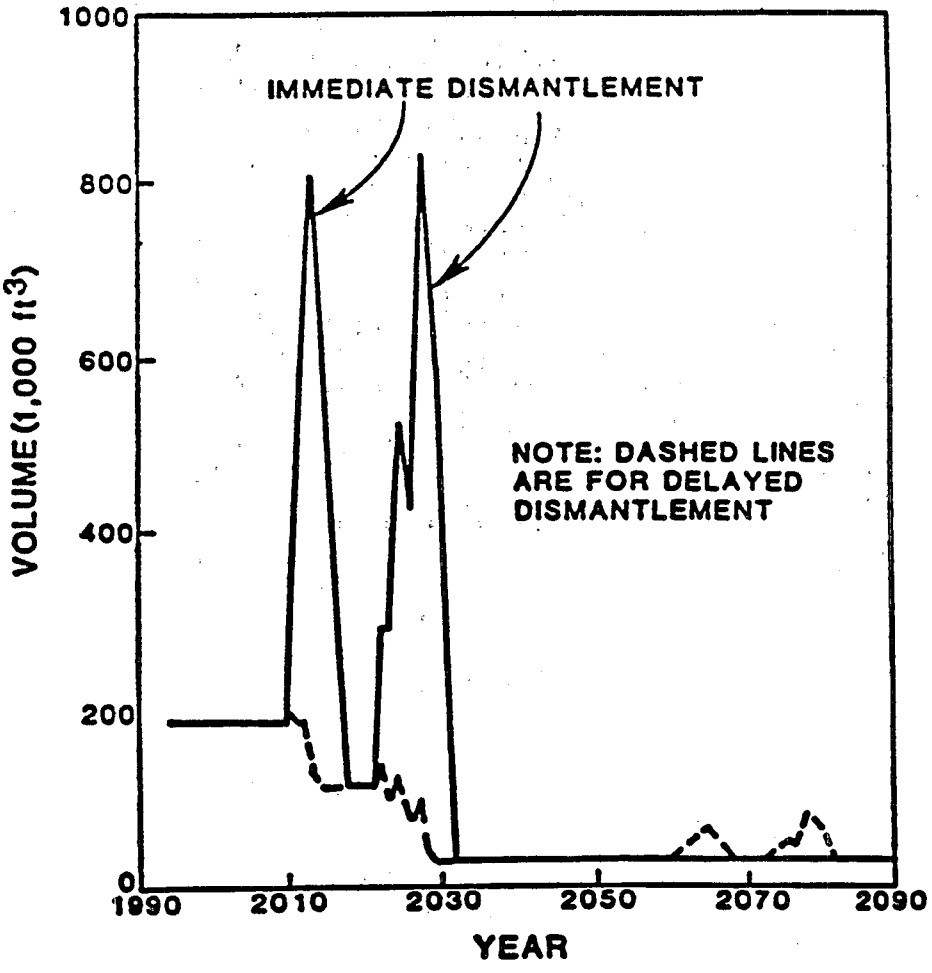

FIGURE 1 IMPACTS OF IMMEDIATE AND DELAYED DISMANTLEMENT OF UTILITY REACTORS ON VOLUMES OF LLW FOR DISPOSAL IN THE CENTRAL INTERSTATE LLRW COMPACT

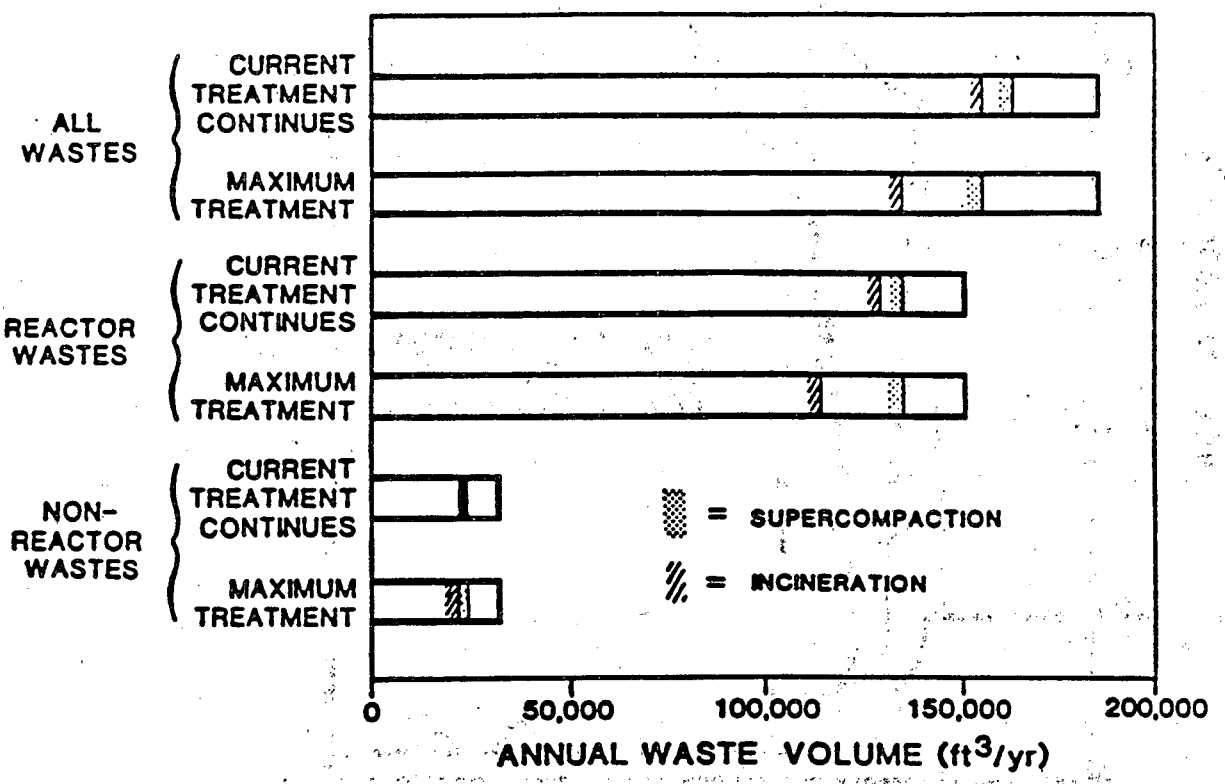

FIGURE 2. POTENTIAL IMPACTS OF REGIONAL TREATMENT FACILITIES.

Bench Scale Studies and Pilot Scale Design of a Modified Volume Reduction-Chemical Extraction System for Radiation Contaminated Soils

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Volume reduction is a subject of interest to everyone associated with the management of radioactive waste. This interest is focused in three general areas. Economics is one area - - - regulatory , transportation, and disposal site factors have during recent years significantly increased the cost to dispose of a cubic yard of low level radioactive waste. From a societal standpoint, waste generators' concerns center on the limited current availability of low level waste burial sites, the uncertainty associated with availability of future State compact sites, and the national policy preferring treatment prior to, or instead of, burial. A second general area of interest is concerned with the broad question of "how clean is clean?". In other words, what exactly are the applicable or relevant and appropriate regulations (referred to as ARARs), should there be one national cleanup standard or should each site have its own standard, and should the standard be related to the specific activity of radionuclides or a dose limit? Thirdly, volume reduction is a generic issue. Every manager of low level radioactive waste, whether in a nuclear power plant, medical institution, Federal facility, or Superfund site, can benefit from the use of volume reduction technology.

The obvious objective of volume reduction is to concentrate the radioactive matrix or separate the non-radioactive from the radioactive components. A wide variety of technologies are in current use, including dryers, compactors, physical separation systems, incinerators, concentrators, evaporators, and chemical systems. Choosing the appropriate volume reduction technology of course depends on the waste form and the strategy for ultimate disposal.

The US Environmental Protection Agency (EPA) Superfund program is directed toward remediation of sites on the National Priority List (NPL). The waste form at many of the NPL sites, that are contaminated with radioactivity, is soil. The only disposal strategy available for radioactive soil is burial. Since the contaminated soil volumes are very large, burial costs will be very high. To investigate technologies that could reduce the volume of soil requiring burial, the US EPA Office of Radiation Programs is conducting a project with the Superfund program that is focused on the remediation of soil contaminated with radium. Factors that must be considered in this soil remediation technology include long-term effectiveness and permanence; reduction

of toxicity, mobility, or volume through treatment; short-term effectiveness; implementability; and cost.

The VORCE (Volume Reduction /Chemical Extraction) project was established to investigate technologies for possible application at the Montclair and Glen Ridge, New Jersey Superfund sites. These two sites were contaminated with ores and ore processing debris from the radium industry that operated in the area in the 1920s. The project has two broad objectives: (1) reduce the activity level of a major portion of the soil to levels protective of public health and return reconstituted soil to the sites, and (2) reduce the volume of contaminated soil by use of physical or chemical techniques that can potentially be applied in residential areas. The project organization consists of two components: Laboratory Studies and Field Treatability.

Laboratory studies are being conducted at the EPA National Air and Radiation Environmental Laboratory in Montgomery, Alabama. Studies include soil characterization protocol development and analysis, evaluation of physical separation techniques (e.g., screening, sieving, magnetic separation), and evaluation of chemical extraction techniques. The radium contaminated soils from the two NPL sites mentioned above have been characterized as to particle size distribution, radioactivity distribution, and mineral and materials composition. Some of the information derived from the characterization studies is shown in Figure 1 for a Montclair soil sample. The laboratory analyses demonstrated that soils from different sites, as expected, exhibit different compositions and radioactivity distributions. For example, the Glen Ridge samples were much higher in activity than Montclair samples, and also had more slag, unprocessed ores, and radium process precipitates. This demonstrates the importance of representatively sampling a given site, and emphasizes the fact that the soil composition and radioactivity distribution dictate the design of the treatment technology . The analyses also showed that the majority of the total activity is in the sand-silt particle size range and that the highest concentrations of radium are in the silt-clay particle size range. The radium in these soils is present as ore materials (e.g., carnotite and uraninite), acid precipitates (radiobarite and amorphous silica), and adsorbed onto materials (slag, quartz, rock, clay minerals).

The initial evaluation of physical separation techniques emphasized screening and sieving. The tests showed that a significant percentage of the radioactivity could be separated into the smaller particle size fractions and that the percentage was highest when a combination wet sieve/vigorous wash technique was used. As shown in Figures 1 and 2 for Montclair soil, the smaller particles are higher in activity (i.e., activity increases as particle size decreases), and there is a dramatic decrease in the activity levels of the sand-gravel fractions for the "Wet Sieve/Vigorous Wash" as compared with the "Dry Screen", corroborating the inverse relationship between particle size and radioactivity. Figures 1 and 2 also show that the small-to-middle sand size (0.15mm to 0.30mm) is the most appropriate particle size range for separation. Laboratory tests demonstrated that 25 - 40% (by weight) of the contaminated soil can be cleaned and returned to the site by use of the wet sieve/vigorous wash technique. The remaining 60 - 75% (by weight) of the soil, consisting of silt, clay, and small sand size particles, would be disposed of.

Some of the components of the contaminated soil have magnetic properties, such as ferruginous particles. The tests to date on magnetic separation demonstrate that the technique has potential application to the soils used in this project. There is evidence that radium contaminated debris was burned in furnaces and the resulting slag and ash materials that were

disposed of in community landfills contained iron compounds. Some of the slag material in the test soils can be removed by a magnetic field, and the laboratory results indicate that the activity of the "wet sieve/vigorous wash" fraction can be further reduced by about five to ten percent. Further tests and evaluations are planned. The expectation is that, provided the laboratory data indicate the technique will be cost-beneficial, a field treatability magnetic separation component will be evaluated.

Chemical extraction techniques are widely used in uranium mining and milling processes. Tests and evaluations are underway at the National Air and Radiation Environmental Laboratory to determine applicability of chemical extraction to the Superfund soils. The objective is to identify acids, salts, or complexing agents, at varying temperatures and concentrations, that potentially could be applied in later stages of the volume reduction system. Initial efforts have been aimed at chemical extraction of radium from the residue from the "wet sieve/vigorous wash" technique, but later tests will investigate application to the unprocessed soil. In addition, techniques for regenerating any acid solutions, salts, or complexing agents used will also be evaluated. Below is an example of the preliminary results from acid extraction, after application of 3M (3 molar) nitric acid at 85 degrees centigrade and a 2.5 to 1.0 liquid to soil ratio.

Radium concentration (picoCuries per gram) in a Glen Ridge soil sample before extraction	Radium concentration (picoCuries per gram) after extraction
60	7
180	13
810	60

Conclusions to date from the laboratory studies can be summarized as follows:

- The "wet sieve/vigorous wash" technique reduced 25 - 40% of the Montclair soil from 180 picoCuries radium per gram of soil to 12 - 15 picoCuries radium per gram of soil.

- The wash water can be recycled (radium is not sufficiently solubilized in the wash water to require removal).

- Reconstituting the soil treated only by the "wet sieve/vigorous wash" technique (12 -15 picoCuries radium per gram) with clean fill will produce 5 - 7 picoCuries radium per gram soil for return to the site, in general conformance with the 40 CFR 192 standards promulgated under authority of the Uranium Mill Tailings Radiation Control Act (UMTRCA) of 1978.

- Preliminary study indicates additional activity reduction may be obtained by magnetic separation and by chemical extraction.

In the Field Treatability/Pilot Scale demonstration phase of the VORCE project, a field treatability unit will be designed based on the results of the laboratory studies, and the unit will be constructed, tested and evaluated. The main objective is to translate the laboratory scale success to the field scale by using off-the-shelf mineral beneficiation equipment, modified as necessary. Equipment components envisioned for use on the VORCE project have been widely

used by the beneficiation industry for many years. These processes and equipment components have however never been used for cleaning soil contaminated with radioactivity. The evaluation phase of the field treatability effort will include thorough field testing of all the equipment and process control parameters.

Figure 3 is a simple mass flow diagram for a VORCE pilot scale field treatability unit. The diagram shows the vigorous wash technique and the potential planned inclusion of the magnetic separation and chemical extraction techniques. Figure 4 is the conceptual design for the vigorous wash technique. This design is briefly described below.

The soil is added onto the "grizzly screen" which removes large sized material (rocks, tree roots, trash). The remaining soil material falls into the "trommel/screen"which provides vigorous mixing, scouring of particle surfaces, and separation and removal of washed gravel size material. The tumbling/cascading action occurring in the pressurized water spray environment scours radioactive material from the gravel- and sand-sized soil particles and breaks down soft soil agglomerates. Scouring removes the majority of radioactive particles adhering to gravel surfaces and a portion of the radioactive particles adhering to sand surfaces. The cleaned gravel size materials are separated from the soil stream by the screen section of the "trommel/screen".

The dirty water containing sand, silt, and disintegrated clay balls passes through the screen openings and into the sloped-bottom water pool of the first stage "classifier". Classification is produced by differential suspension whereby settled material is carried along the inclined bottom and discharged at the open end of the trough. Vigorous washing of this settled material is produced by the tumbling/rolling action of the rotating spiral ribbon which grinds sand grain against sand grain and thereby removes deleterious material. In differential suspension, rounded, denser, larger soil particles tend to sink faster than angular, less dense, smaller soil particles. The latter particles tend to remain suspended near the surface of the water pool and thereby overflow the weir. The "first stage classifier" therefore produces two product streams:

a. Material which has settled to the bottom of the water pool, traveled the length of the rotating screw to the upper end of the trough, and has been discharged into the input of the "attrition mill".

b. Material which has remained in suspension in the water pool and follows the water overflowing the weir into the input of the "sloped plate clarifier". This material will be greatly enhanced in radioactivity.

The material discharged at the upper end of the "first stage classifier" receives prolonged agitation washing in the "attrition mill". The "attrition mill" is composed of two cells, each containing a high speed turbine shaft with two propellers set at opposing pitches. The propellers force the material grains to abrade each other while carried in water suspension within the two cells, thereby scouring superficial adhesions and coatings and suspending them. These newly suspended, small particle size adhesions and coatings will be enhanced in radioactivity and the coarser particles from which they were removed will be cleaned of radioactivity. The output from the "attrition mill" is discharged to the water pool of the "second stage classifier".

The "second stage classifier" provides the final washing and particle size separation. The "second stage classifier" produces two product streams:

a. Material which has settled to the bottom of the water pool, traveled the length of the rotating screw to the upper end of the trough, and discharged onto a stockpile. This material will have been cleaned by the washing, scrubbing, scouring action of the "trommel/screen", "first stage classifier", "attrition mill", and second stage classifier", and may be returned to the original excavation site.

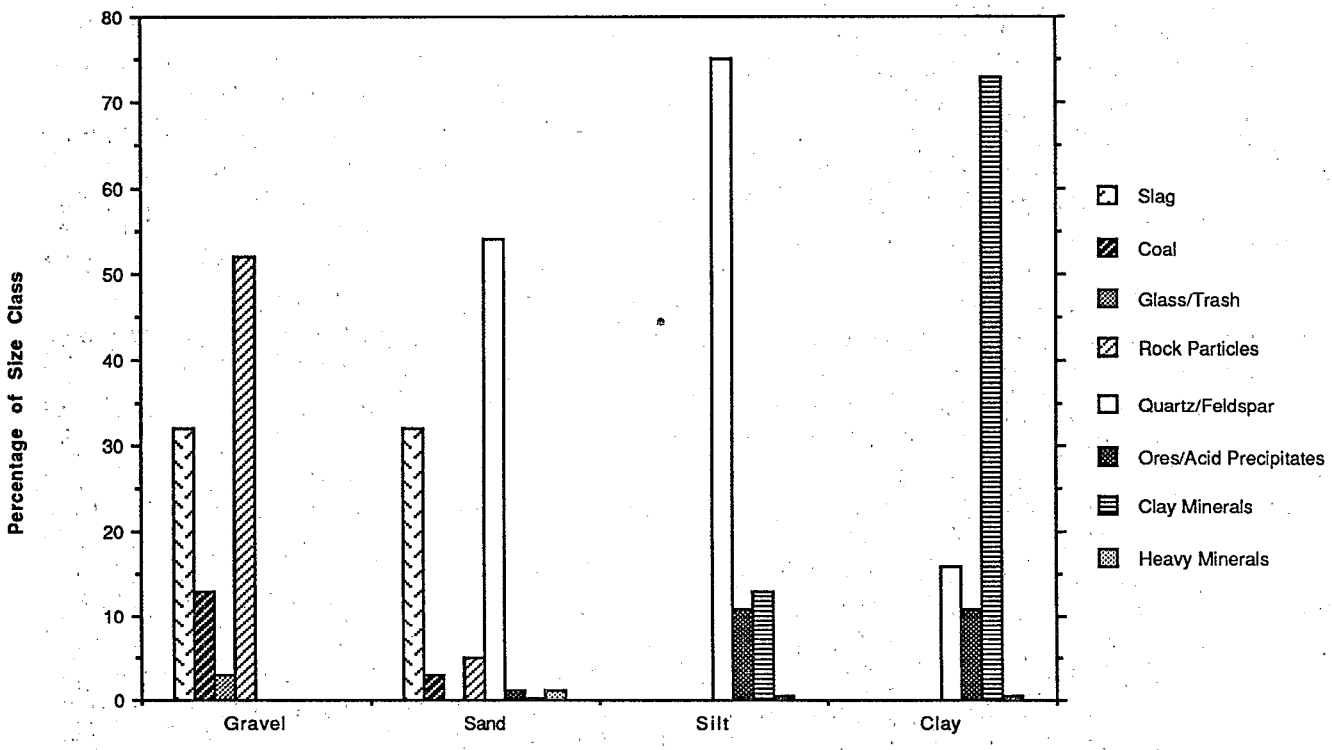
b. Material which has remained in suspension in the water pool and follows the water overflowing the weir into the input of the "sloped plate clarifier". This material will be greatly enhanced in radioactivity.

The "clarifier" separates solids from water and produces a sludge, containing the radioactive materials that have been removed from the original contaminated soil. The "plate and frame filter press" removes the majority of the water from the clarifier sludge, thereby allowing the water to be recycled. The solid filter cake produced by the "plate and frame filter press" contains the radioactivity removed from the original contaminated soil and should be disposed of as radioactive waste.

Figure 5 shows a general layout of the VORCE field treatability pilot scale unit. The treated soil stream from the trommel and from the second stage classifier will be sampled and analyzed to determine the radium concentration. Soil that meets the cleanup requirement will be moved to the treated soil pile for mixing with fill soil. The resulting reconstituted soil will be returned to the site. If, because of unusually high activity input soil, the streams are not sufficiently clean, they will be moved to the reject soil pile for recirculation through the system up to three additional cycles. The filter cake will be analyzed and packaged for disposal.

To summarize, laboratory scale tests and evaluations of techniques for reducing the volume of radium contaminated soil for disposal have demonstrated that a significant portion of the contaminated soil can be cleaned by a vigorous water wash technique and returned to the excavation site. The laboratory work has also demonstrated the importance of soil characterization, and has provided preliminary results that indicate magnetic separation and chemical extraction techniques can provide additional volume reductions. A conceptual design for a field treatability pilot scale unit to duplicate the laboratory results has been completed. The design relies on readily available components. Construction, test, and evaluation of the unit is planned.

Figure 1: Soil Characterization Data (Example)
(180 picoCuries Ra226/gram Montclair soil)



Size Class → decreasing particle size				
26	53	20	1	% (volume or weight) of Total Soil
44	98	396	688	Average pCi Ra226/gram
8	35	53	5	% of Total Soil Ra226

Figure 2: Wet Sieve/Vigorous Wash Data (Example)
(180 picoCuries Ra226/gram Montclair soil)

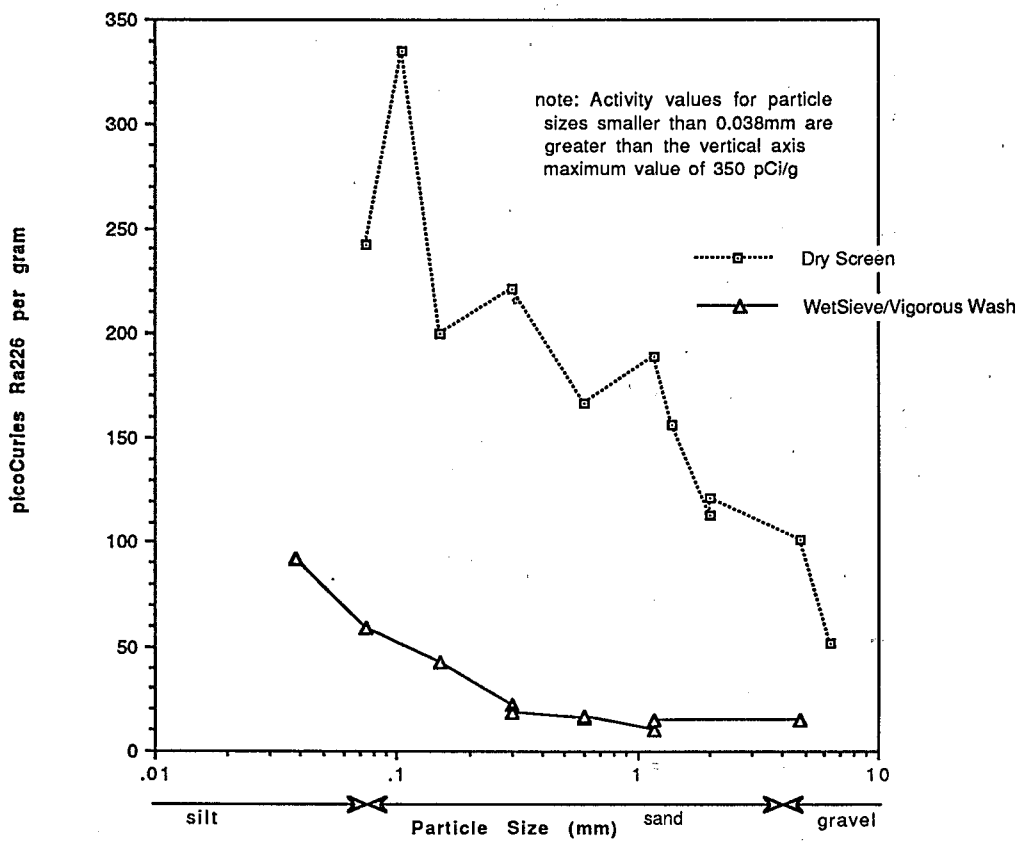


Figure 3: VORCE Flow Diagram

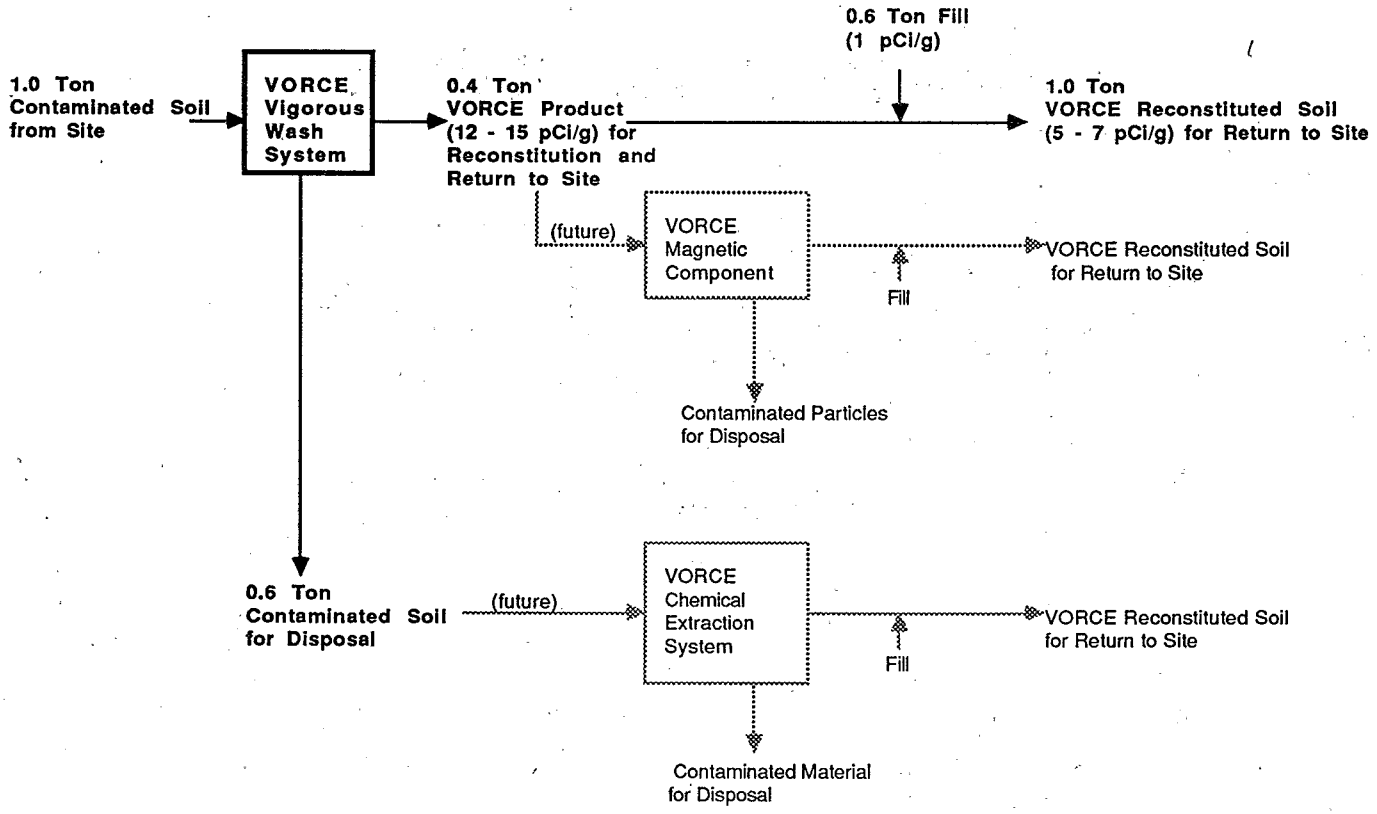


Figure 4: Conceptual Design

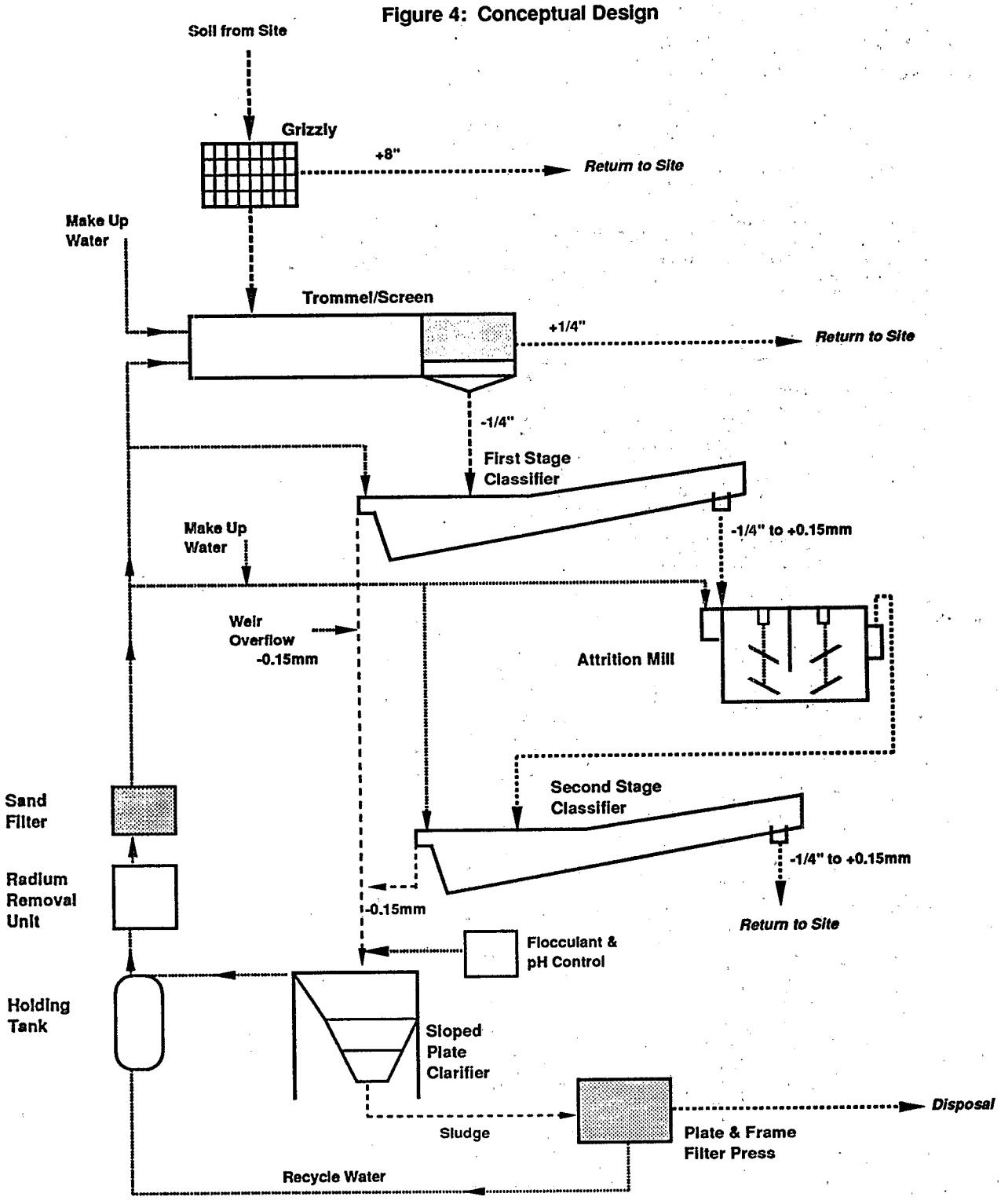
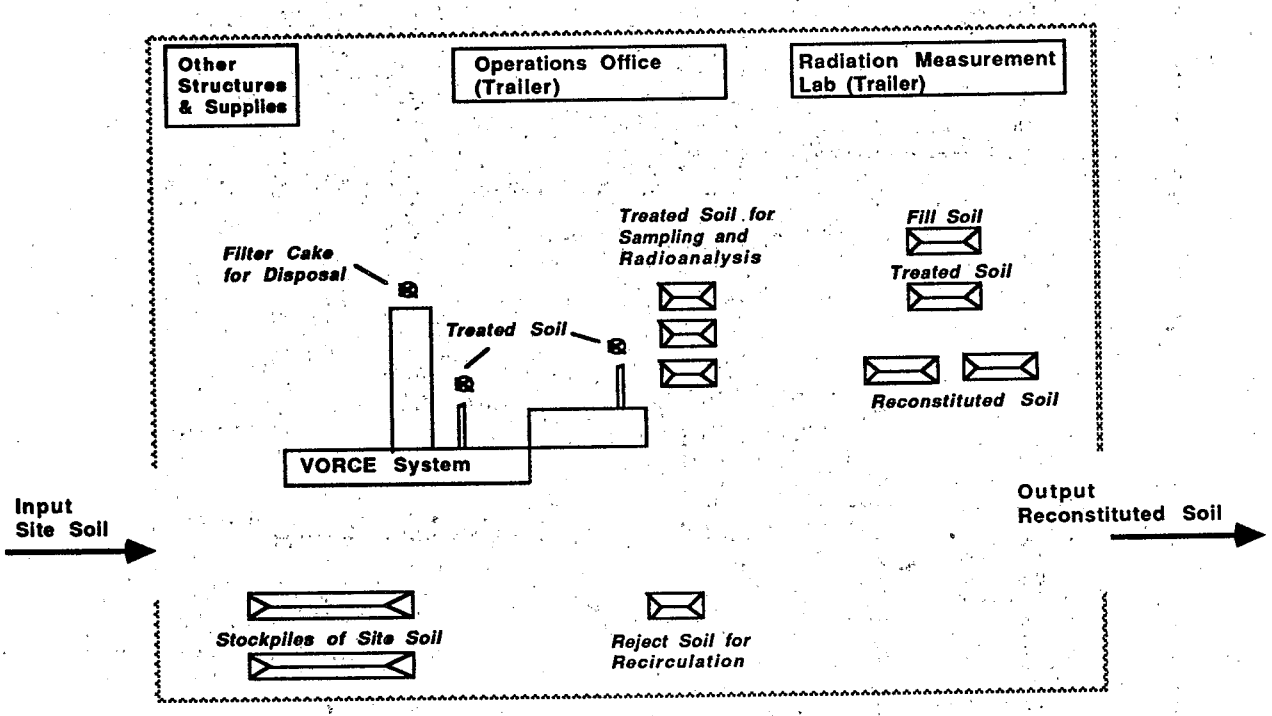


Figure 5: Plant Layout



*Residual Radioactivity Cost Impact Evaluation**

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ABSTRACT

Studies are in progress at Pacific Northwest Laboratory to evaluate the cost differentials (impacts) for compliance with different residual radioactivity limits governing the release of land and structures for unrestricted use following the decommissioning of licensed nuclear facilities. The initial work has focused on the decommissioning of a reference uranium fuel fabrication facility to achieve alternative effective dose equivalent levels ranging from 80 to 1 mrem/yr for the maximally exposed individual. Concrete and soil removal and disposal costs were calculated for the reference uranium fuel fabrication facility as a function of these target levels. The disposal cost evaluation considered both low-level waste (LLW) disposal and the effect of establishment of a below regulatory concern (BRC) disposal option based on compliance with a 1, 4 or 10 mrem/yr individual dose level. The results indicate that removal+disposal costs for concrete structures are relatively insensitive to decreasing target levels for the LLW disposal option, but costs decrease at the lower target levels for the BRC waste disposal option. The availability of the BRC waste disposal option favors demolition over the preservation of decommissioned structures. For contaminated soil, the removal+disposal costs increase with decreasing target levels for the LLW disposal option, but similarly can decrease at the lower target levels for the BRC waste disposal option.

INTRODUCTION

The preliminary studies described in this paper used a reference uranium fuel fabrication facility decommissioning study (Elder and Blahnik 1980) to develop a cost analysis methodology and illustrate its application to the evaluation of decommissioning cost impacts.

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The analysis was limited to the land and structures remaining at the completion of equipment dismantling and building decontamination operations, and assumed a starting dose of 100 mrem/yr. Based on these guidelines, concrete and soil removal and disposal costs were calculated for representative decontamination and decommissioning (D&D) operations capable of achieving alternative (target) residual radioactive contamination levels of 80 to 1 mrem/yr for the maximally exposed individual. Low-level waste (LLW) disposal costs were calculated based on shipment of the LLW to a regional waste compact site. The effect of the establishment of a below regulatory concern (BRC) waste disposal option based on compliance with a 1, 4 or 10 mrem/yr individual dose level also was evaluated.

The reference facility plant area selected to provide a conservative case for the impact evaluation was an 8 x 8 x 5 m ground floor room with reinforced concrete floor, concrete block walls and reinforced concrete ceiling. The site area was representative soil containing radionuclides deposited from normal plant operations. The physical condition of the reference facility at the conclusion of the decommissioning study (starting point of this evaluation) was defined by reviewing the proposed decommissioning operations as specified in Elder and Blahnik (1980), supplemented by discussions with participants in the original study. Decommissioning of the plant and site areas would be effected by:

1. Disassembling, decontaminating, packaging and disposing of all equipment.

2. Removing all remaining piping, electrical conduit, and local filter systems and ducting. The main ventilation ducts were left in place since they were protected by HEPA filters on the equipment and room ducting.

3. Decontaminating the floors, walls and ceilings with detergent cleanser, powered brushes, and hand wipes. The floor tiles and strippable seal coatings would be removed.

4. Removing hot spots in the floors and walls by chipping and vacuuming.

5. Removing all inside and outside piping from the pipe trenches, along with any hot spots in the trenches.

6. Removing contaminated material and liners from the waste treatment lagoons, along with any contaminated soil beneath the liners.

The reference decommissioning operation thus would leave the buildings intact, with all contaminated equipment and fixtures removed and the interior surfaces decontaminated. There would be no general site decontamination, just selective removal of hot spots associated with pads, piping and the lagoons.

COST EVALUATION

Unlike other facilities, the fractional activities for a uranium fuel fabrication plant are not location or process specific; i.e., the residual radioactive mixture is the same for all structure and site surfaces. However, the initial contamination levels and the concentration profiles can vary

significantly as a function of the handling+processing step, the form of the contamination (gaseous, liquid, powder), and the type and condition of the surface (metal, concrete, porous, sealed, etc.).

The radionuclide inventory estimates in Elder and Blahnik (1980) addressed the equipment removed during the decommissioning operation, but not the structure and site surfaces except for the lagoons. A survey of recent decontamination, decommissioning and radiological survey literature was conducted to provide some guidance on post-decommissioning contamination levels and profiles and also to obtain current D&D cost, worker dose and waste volume data. The only residual contamination information relevant to the uranium fuel fabrication plant reference case was some limited soil contamination data from the post-remedial-action radiological survey of a decommissioned mixed oxide fuel development and fabrication facility (Flynn et al. 1984). All uranium and thorium readings were within the range of normally expected background concentrations except for one area with isotopic ratios indicative of a release or spillage of enriched uranium. The contaminated zone was subsurface (5-10 cm depth), with an average radionuclide concentration of 6.4 pCi/g.

This lack of specific data or estimates defining the initial residual concentration values for the reference facility is not critical since the cost and worker dose impacts are evaluated using differences rather than absolute values. The initial concentration values were therefore calculated based on an assumed starting dose of 100 mrem/yr. It was further assumed that the plant walls and ceilings contain 50% and 10%, respectively, of the corresponding floor contamination inventory.

A concentration versus depth functional relationship must be established for each structure and site area in order to estimate the effectiveness of the applied decontamination processes and the associated costs and worker exposure as a function of the amount of material removed. The literature survey cited earlier identified several different types of observed contamination distribution profiles reflecting the origin and subsequent history of the various possible surface-contaminant interactions, i.e., contamination incorporated in surface coatings, diffusion or mixing into the base material, previous decontamination operations, subsequent coverage with paint or clean soil, contaminated versus activated material, etc. No concentration profile information directly relevant to the uranium fuel fabrication plant reference case was identified other than the previously noted example of subsurface soil contamination.

In the absence of applicable data, diffusion theory was used to develop a generic concentration versus depth relationship. The concentration profile for a substance deposited on a surface and allowed to diffuse into the base material for a time t is given by the exponential expression: $C = \alpha \exp(-x^2/\beta t)$, where C is the concentration at the depth x. Solving a diffusion problem involving a series of line sources or a single replenished source yields an expression of the form: $C = \alpha \operatorname{erfc}(x/\beta t)$, where erfc is the error-function complement. This expression was selected as the best representation of a generic concentration profile, since the contamination diffused into a surface could represent the cumulative effect of multiple sources of varying strength applied at different times and for variable time intervals.

The required relationship between removal depth and the percent of the total contamination inventory remaining after a decontamination operation was derived by integrating the error function. An adjustable scaling factor was used to define the depth corresponding to

a specified percent removed or remaining. Figure 1, for example, illustrates the calculated depth profile curves for a 50% penetration depth of 2 mm, 1 cm, and 3 cm from the original deposition surface. The same basic contamination inventory versus depth relationship thus can be used to represent contamination that is predominantly near the surface or that has penetrated a significant distance into concrete or soil.

Once the concentration profile and corresponding removal depths have been defined, the specific decontamination and dismantlement processes applicable to each of the major surface types (metal, concrete, soil, asphalt, etc.) can be selected. The major selection considerations for decontaminating low activity surfaces are 1) required removal depth, 2) access constraints, 3) contamination control and recontamination potential, 4) waste type and volume, and 5) cost. The following are the candidate decontamination and dismantlement processes for concrete and soil for the reference uranium fuel fabrication facility:

Concrete Decon. (0-1 cm):	Scabbler
Concrete Decon. (1-5 cm):	Spaller
Concrete Demolition:	Backhoe-Mounted Ram
Soil Removal:	Modified Backhoe

The values in parenthesis are the optimum removal depth ranges for the scabbler and the spaller. Demolition of walls, floors, etc., is required if the amount of material removed impairs structural integrity. For this evaluation, the demolition criteria are 1) removal of more than one-half of the 10-cm floor thickness, 2) removal of more than one-fourth of the 20-cm wall or 10-cm ceiling thickness, since these would be decontaminated on both surfaces, and 3) automatic demolition of the ceiling along with the walls.

The decontamination costs and associated waste volumes for the reference fuel fabrication facility structure and site surfaces were calculated using the following steps and associated assumptions:

Concrete Decontamination

1. Calculate the concentration profiles for the defined surfaces and an assumed 50% penetration depth using the generic concentration profile derived previously and initial inventory values of 1700 $\mu Ci/m^2$ for the floor, 850 $\mu Ci/m^2$ for the walls, and 170 $\mu Ci/m^2$ for the ceiling. The penetration depths, reflecting the volume containing 50% of the total inventory as measured from the original deposition surface, are assumed to range from 0.2 cm to 5 cm.

2. Calculate the total amount of material that must be removed for each surface to reduce the dose to the specified target level. These calculations are based on the inventory factors (total $\mu Ci/m^2$ of surface) and the specified surface activity factors (dpm/100 cm^2 of surface). It is assumed that the higher concentration in the floor is removed first, and that the walls and ceiling are decontaminated only when their dose contribution becomes significant. Since there may be substantial additional costs or technical difficulties associated with the precise establishment of the required removal depth, the calculated depth is increased by 50% to reflect the uncertainties inherent in an actual decommissioning operation.

3. Calculate the LLW/BRC waste boundary for each surface to establish the required removal depths for each waste type. The radionuclide concentration defining BRC concrete waste from the reference uranium fuel fabrication facility is assumed to be 20 times the allowable residual concentration for the same target dose levels. For example, BRC waste at the 4 mrem/yr BRC level has the same average radionuclide concentration as the residual concrete for an 80 mrem/yr target dose level. The LLW/BRC waste boundary is established by starting at the maximum residual contamination removal depth and then calculating the average radionuclide concentration for the removed material until the BRC limit is reached. The calculated LLW removal depth is increased by 50% to reflect uncertainties in establishing this boundary in an actual decommissioning operation.

4. Specify, for each removal depth, the appropriate decontamination or dismantlement technique and calculate the number of applications required to remove material to at least 80% of the specified depth. Use of the 80% value rather than the full depth minimizes excessive waste generation just to remove a small remaining fraction of the required inventory, and is consistent with the field uncertainties in removal depths and the amount of material actually removed with each technique.

5. Calculate the decontamination cost to reach each target residual radioactivity level starting from the initial conditions.

Soil Decontamination

1. Calculate the concentration profile for soil using the generic concentration profile derived previously and an initial inventory value of 210 μCi/m2 for soil homogeneously contaminated to a depth of 15 cm, and to twice this value for contaminated soil covered with 15 cm of clean soil.

2. Calculate the total amount of soil that must be removed to reduce the dose to the specified target level, assuming no subsequent coverage with clean soil. As with the concrete surfaces, the calculated depth is increased by 50% to reflect the uncertainties inherent in an actual decommissioning operation.

3. Calculate the LLW/BRC waste boundary, the decontamination costs, and the disposal volumes using the procedures outlined for concrete. The radionuclide concentration defining BRC waste from the reference uranium fuel fabrication facility site is assumed to be four times the allowable residual concentration for the same target dose levels.

Waste Handling and Disposal

It is assumed 1) that all removed concrete and soil that is classified as LLW is sent to a generic regional waste compact site for final disposal, 2) the disposal volume for the scabbled, spalled or demolished concrete is ~1.4 times the removal volume and 3) the soil disposal volume is equal to its removal volume.

A representative distance of 390 miles from the reference facility to the disposal site was established by averaging the one-way distance from nuclear power stations to the existing

disposal sites. A truck transportation cost of ~$100/m^3 of removed material for concrete with a density of 2300 kg/m^3 was calculated by assuming a representative one-way charge of $2.25/mile and a weight limit of 20,000 kg. The corresponding transportation cost for the 1500 kg/m^3 soil is $70/m^3.

Current LLW disposal charges for the three existing disposal sites were averaged to give a disposal cost of $1500/m^3 for the removed concrete and $1100/m^3 for the soil. Adding packaging costs of $200/m^3 for the concrete and $130/m^3 for the soil gives a total waste handling+disposal cost of $1800/m^3 of removed concrete and $1300/m^3 of excavated soil.

It is further assumed that any removed concrete and soil that can be classified as below regulatory concern waste is sent to a local sanitary landfill for final disposal. Other assumptions are 1) the waste is transported 30 miles at a cost of $17/1000 kg, or $40/m^3 of removed material for the concrete and $25/m^3 for the soil, and 2) the disposal charge is $26/1000 kg, or $60/m^3 of removed material for the concrete and $40/m^3 for the soil.

This gives a total BRC waste handling and disposal cost of $100/m^3 of removed concrete and $65/m^3 of excavated soil, or almost a factor of 20 less than the corresponding direct cost for LLW handling and disposal. However, these savings are at least partially offset by increased costs for monitoring, verification and certification of the BRC waste.

RESULTS AND DISCUSSION

The following are the key cost impact evaluation results for the reference concrete room and site area. The associated figures show the calculated concrete or soil removal and waste disposal costs required to go from the starting condition to a surface contamination level less than the prescribed limit. The removal and disposal costs are displayed as separate bands that sum to the total cost to illustrate the variation of the individual cost elements and the total cost with the target levels. The cost differential or "impact" of decreasing the target levels is reflected by the slope of the upper boundary, or total cost line. It must be emphasized that these cost figures do not include front end and indirect but substantial costs such as management, planning, engineering, training, radiological support, and safety because they would be incurred for any target level requiring significant additional decontamination.

Concrete Decontamination

With all removed material classified as LLW, the concrete room removal+disposal costs increase substantially with increasing contamination penetration depth (Figures 2 and 3) at the lower target levels. However, the total cost is either constant or changes very little with residual contamination level for target levels below 30 mrem/yr at any particular assumed penetration depth. For contamination concentrated near the surface (Figure 2), the total cost is quite insensitive to decreases in the target level. This reflects the discrete rather than continuous nature of the concrete removal processes and the ability to remove all of the contamination with one application. In the case of the reference room and the 0.2-cm penetration depth, for example, the amount of material removed by one application of the spaller (2.5 cm) is more than sufficient to reach any specified target level including 1 mrem/yr.

For the same contamination inventory distributed over a depth of several centimeters (Figure 3), the removal+disposal costs are comparatively low at the highest target levels, and then increase sharply to a maximum LLW disposal cost reflecting demolition of the reference room. The cost is low initially because of the reduced concentration levels associated with the deeper penetration of the contamination. However, once the target level decreases sufficiently to require significant decontamination, the amount of material that must be removed requires demolition of the structure.

The preceding analyses were based on the classification and disposal of all removed concrete as low-level waste at $\$1800/m^3$. An alternative option is to classify removed material complying with a 1, 4 or 10 mrem/yr individual dose level as BRC waste that can go to a local sanitary landfill at a disposal cost of $\$100/m^3$. The effect of this BRC option on removal+disposal costs as a function of target level is quite dramatic as illustrated in Figure 4 for an assumed BRC level of 10 mrem/yr. The removal+disposal costs at the highest target levels are the same. The material nearest the surface has the highest contamination concentration, and is classified as LLW regardless of the available disposal options.

As the target levels decrease, however, lower concentration material is removed that meets the BRC criterion. The total cost decreases rather than increases at the lower target levels since demolition is a comparatively low-cost operation and the resulting large waste volumes do not have a corresponding large disposal cost. In addition, if the definition of BRC waste is based on an average contamination concentration, the volume of LLW and the attendant removal+disposal costs will decrease rather than increase at the lower target levels. This results from an effective dilution of the original near-surface LLW by lower concentration material from deeper in the structure; i.e., the volume of BRC waste increases at the expense of the LLW.

This LLW cost decrease can be converted to a constant LLW removal+disposal cost by defining BRC waste based on a maximum rather than an average contamination concentration as illustrated in Figure 5. However, the total cost is still lowest at the lower target levels where demolition is required. Since most of the BRC cost for intermediate target levels is for removal operations to preserve the structure, a further cost reduction is possible by choosing to demolish the structure regardless of the target level. The availability of a BRC waste disposal option thus could favor demolition over preservation for decommissioned structures since the removal and disposal costs would be the lowest of any option and almost independent of the residual level.

Soil Decontamination

The soil LLW removal+disposal costs for contamination concentrated within the first few centimeters of the surface are independent of the target level. Unlike the concrete case, the soil LLW removal+disposal costs increase significantly with decreasing target levels for deeper assumed contamination penetration values as shown in Figure 6. However, the availability of a BRC waste disposal option for soil results in a moderation, or even a decrease, in total removal+disposal costs at the lower residual contamination levels (Figure 7), but this beneficial effect occurs at lower target levels for soil than for concrete.

These results for the uranium fuel fabrication facility reference room and site area provide valuable insights regarding key factors that can affect the cost impacts for compliance with different residual radioactive contamination levels. However, this evaluation is quite limited,

and the results and conclusions must be regarded as preliminary. Additional work is needed to address the number of rooms and site areas that are contaminated, the effect of the differing initial contamination levels for these areas, and the impact of the cost variations for these specific decontamination operations on the total decommissioning cost.

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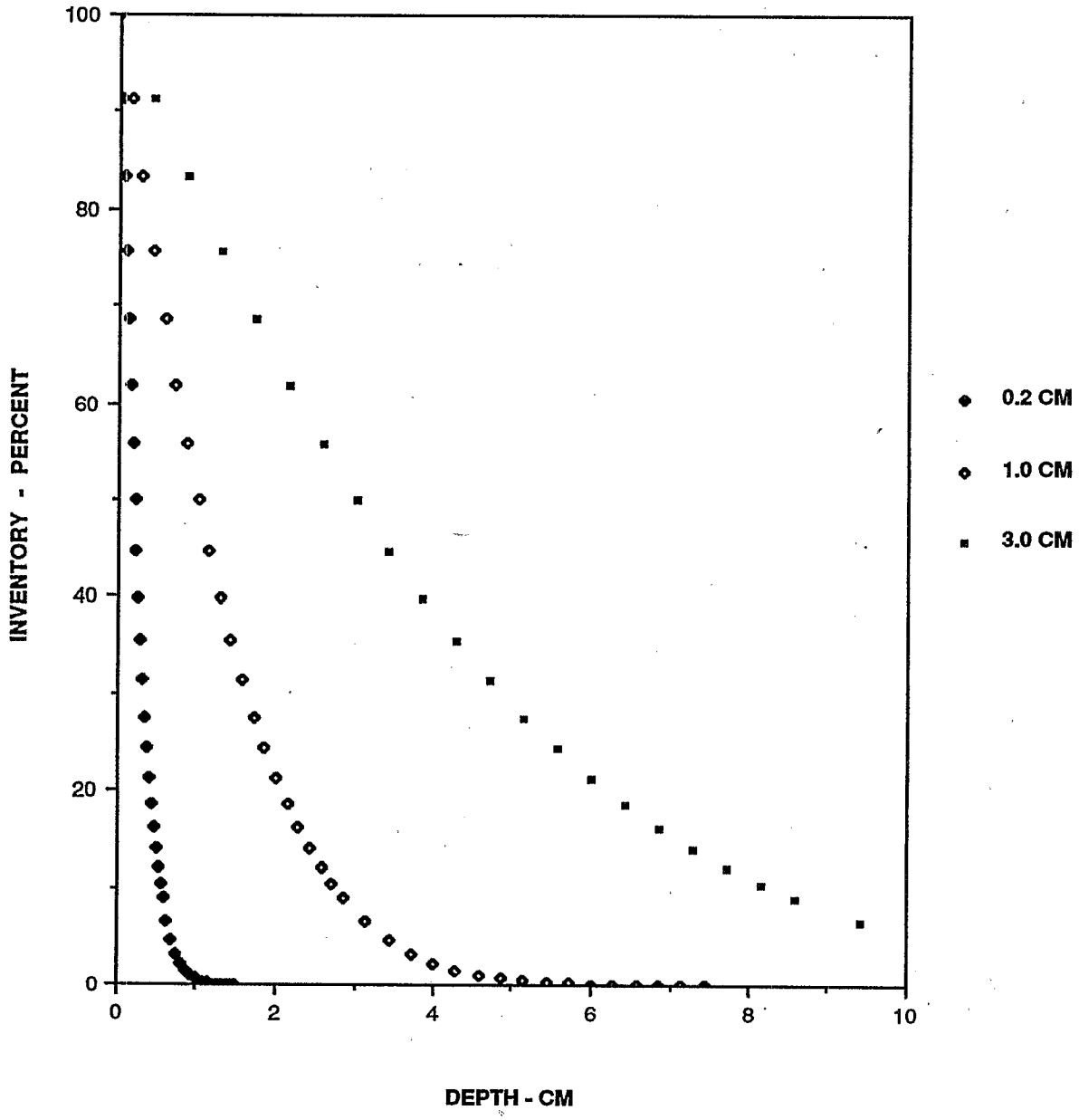

FIGURE 1. Calculated depth profile curves

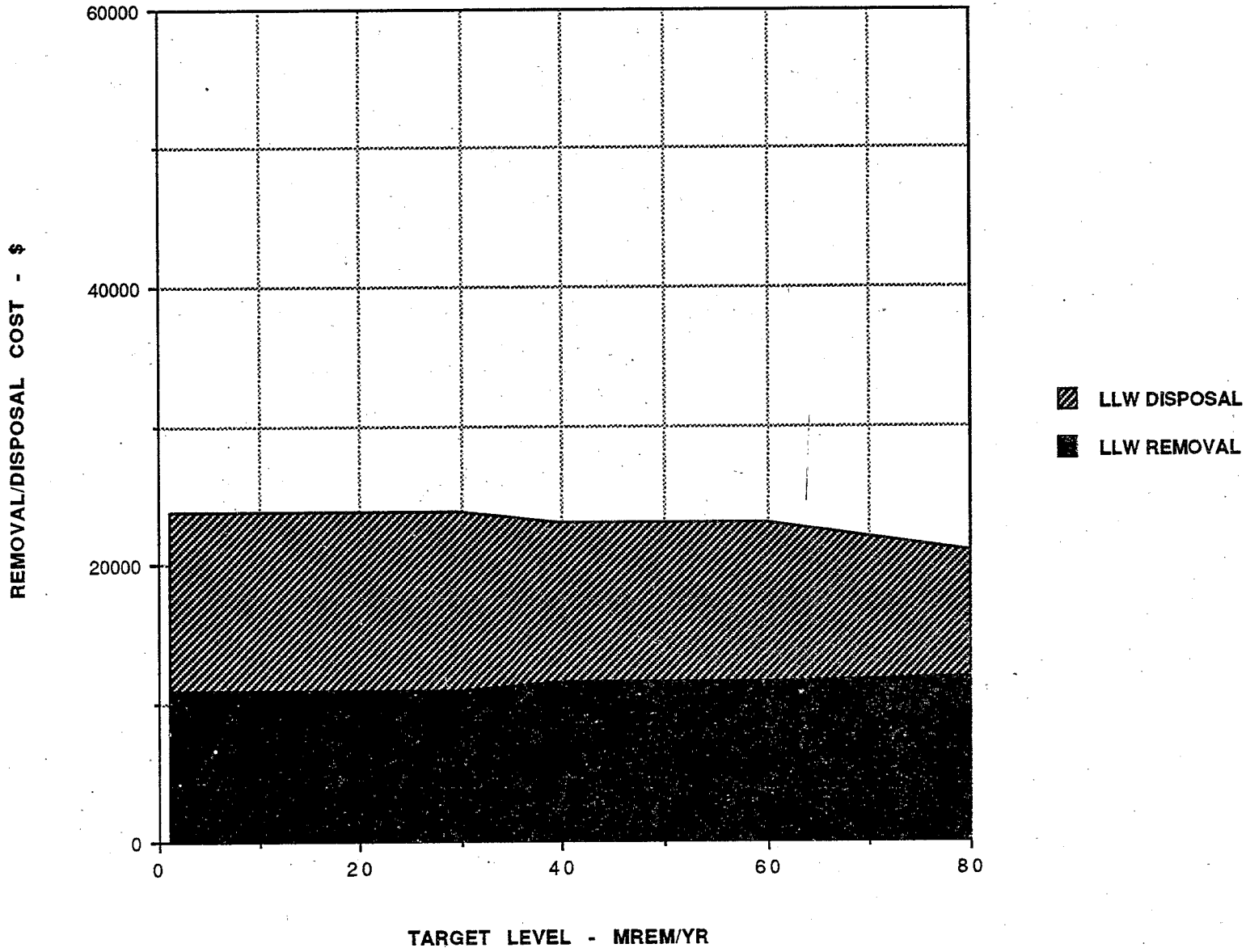

FIGURE 2. Concrete removal + disposal costs for a 0.2-cm penetration depth

FIGURE 3. Concrete removal + disposal costs for a 5.0-cm penetration depth

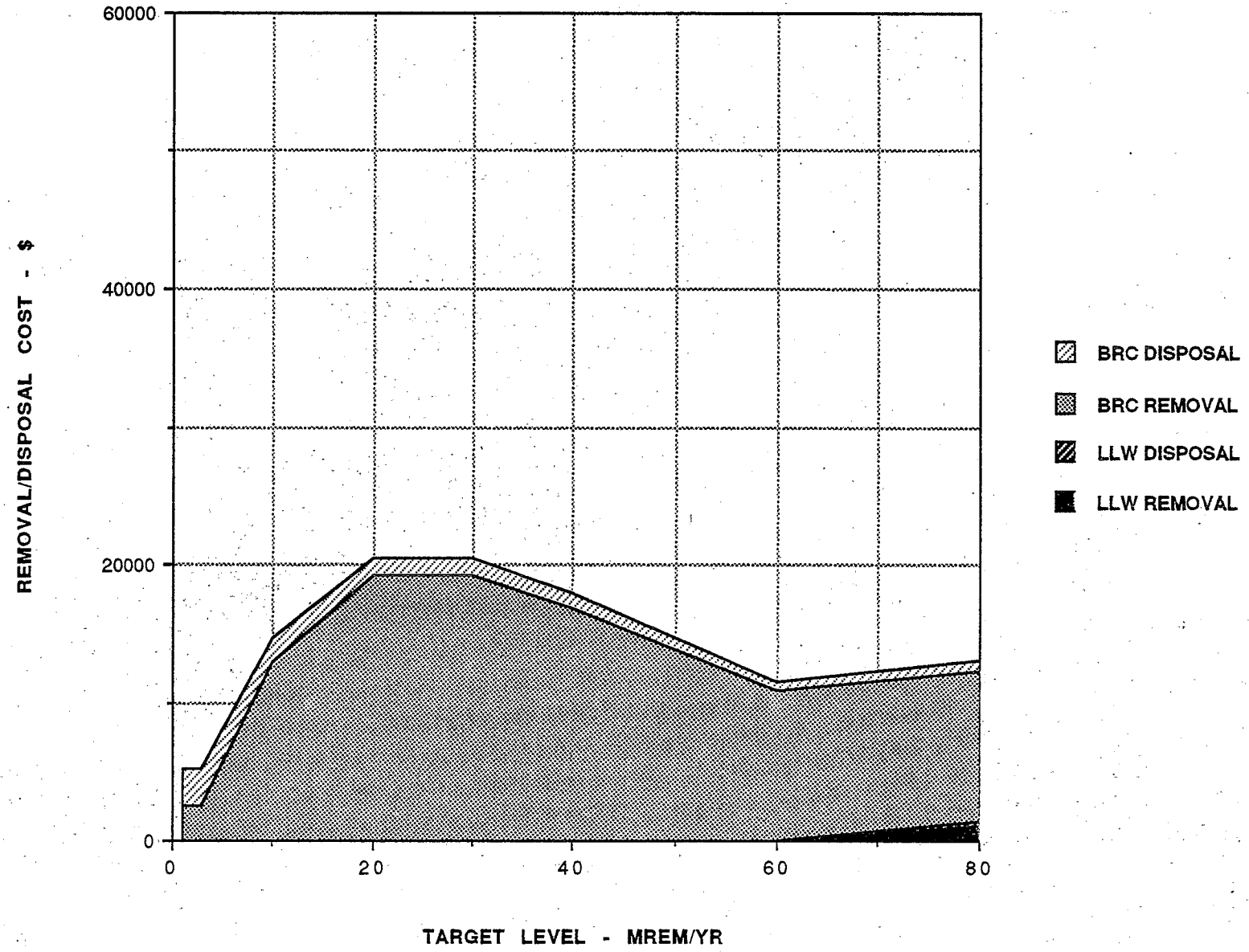

FIGURE 4. Concrete removal + disposal costs for a 10.0 mrem/yr average BRC level

FIGURE 5. Concrete removal + disposal costs for a 10.0 mrem/yr maximum BRC level

FIGURE 6. Soil removal + disposal costs for a 10.0-cm penetration depth

FIGURE 7. Soil removal + disposal costs for a 10.0 mrem/yr average BRC level

Session III

Health Effects

Experience in Decontamination and Reuse of the Large-Scale Radiochemical Laboratory and the Research Reactor at the Japan Atomic Energy Research Institute

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ABSTRACT

This paper reviews, from a health physics viewpoint, some of our experience in decontamination and reuse of a heavy-water cooled research reactor and a large-scale radiochemical laboratory at the Japan Atomic Energy Research Institute. The research reactor was decommissioned with plans for its reactor containment to be reused for a new reactor. Residual radioactivity particular to this case was tritium remaining in concrete matrices of the reactor containment. The primary health risk possibly encountered in reusing this type of facility would be caused by inhalation of gaseous tritium released from concrete. The radiochemical laboratory was decontaminated and reused as office rooms. The health risk from reusing this facility would result from both internal and external exposure to various forms of radioactivity.

Generalizing from this experience, the information needed for estimating the risks in these types of facilities is examined. Possible forms of residual radioactive criteria for each of these cases are discussed.

INTRODUCTION

More than 30 years have passed since the establishment of the Tokai Research Establishment of the Japan Atomic Energy Research Institute (JAERI). Therefore, in the Establishment, there are some old facilities which have lost their usefulness or flexibility to users' expanding requirements.

Typical examples of these timeworn facilities at JAERI were the Japan Research Reactor No. 3 (JRR-3), and a large-scale radiochemical laboratory called the Research Laboratory Building No. 1. These two facilities needed to be repaired, to fill new needs.

In this paper, decontamination experiences with these two facilities and their reuse are discussed from a health-physics viewpoint. In addition, some characteristics of health risks, and data needed for estimation of the risks due to residual radioactivity in these types of facilities, are pointed out.

REUSE OF BUILDINGS

Objects for decontamination or reuse can be categorized into "sites", "buildings" and "materials". The reuse of "buildings" is discussed in this paper. The two "buildings" discussed here can be classified as a heavy-water cooled research reactor in the "reactor category" (test reactors, research reactors, commercial power plants, reactors in ships) and a chemical laboratory in the "laboratory category" (physics laboratories, including accelerators, chemical laboratories), respectively. The Japan Research Reactor is being prepared for "restricted reuse", and the Research Laboratory Building No. 1 for "unrestricted reuse".

JRR-3

JRR-3 (10 MW), the first reactor designed, manufactured and installed with only Japanese domestic techniques, was operated from 1962 to 1983 in the Tokai Research Establishment. Experiments in design, construction, and operation were performed with it for developing and testing reactors. It also produced radioisotopes used in various fields of industry and science. The reactor fuels were natural and 1.5% enriched uranium oxides, and the coolant was heavy-water.

During its 20 year operation, users of the reactor carried out increasingly complicated experiments, and required a higher quality of performance of the reactor. Eventually, it could no longer satisfy the users' needs sufficiently.

The dismantlement program planned to remove the reactor block without taking apart the reactor containment, and to construct a new reactor with higher power and efficiency inside the same containment. The reactor containment consists of reinforced concrete and is 33 m in diameter, 27 m in height, and cylindrical in shape. It should be emphasized that reuse is intended to be restricted, and that the inside of the reactor containment will be treated as a "radiation controlled area" as before.

The one-piece removal of the 2,250 ton reactor block attracted much public attention. The duration of the dismantlement continued from August 1984 to March 1987.

Types and Levels of Residual Radioactivities

When reactor JRR-3 ceased its operation, a considerable amount of radionuclides remained in its installation and equipment. Most of the residual contamination was removed with the reactor block. However, tritium remained in the concrete walls, floors, and ceilings of the containment.

The tritium contamination of the reactor containment was due to the heavy-water coolant. The heavy-water was activated and tritium was generated and stored in the coolant system. The

tritium concentration in the coolant was about 40 MBq per cubic centimeter in 1984, after about 20 years of operation.

The gaseous heavy-water containing the tritium diffused into the reactor containment, deposited onto its surfaces, and penetrated into the concrete structures. The heavy-water released from the coolant system onto floor surfaces during maintenance work directly penetrated concrete structures, and evaporated and then contaminated other floors, walls and ceilings.

Figure 1 shows an example of tritium concentration distribution in the concrete of the reactor containment. Widespread tritium contamination in the reactor containment should be noted. The tritium concentration levels in the concrete samples of the basement floor, i.e., around the coolant system, were higher and more scattered than on the first floor. The basement floor was probably contaminated directly with tritiated heavy-water released from the coolant system, while the first floor was contaminated by tritium vapor. Typical profiles of tritium concentration in the concrete of floors and a wall are shown in Figure 2. The tritium penetrated about 60 cm into floors and walls of JRR-3.

The tritium in the concrete, released to the atmosphere of the reactor containment, could generate airborne contamination. The tritium concentration in the atmosphere of the reactor containment was measured before the removal of the primary coolant system. The maximum concentration was approximately 0.7 Bq per cubic centimeter of air in 1979. This value decreased to about 70 micro-Bq per cubic centimeter of air after removal of the reactor block. The latter level of the tritium concentration is considered to be caused by the release of residual tritium in the concrete structures inside the reactor containment.

Decontamination

The decontamination process, which aimed to reuse the reactor containment, involved surveys of contamination level, the decontamination itself, and measurements of the level of residual radioactivity afterwards. Although various types of decontamination were performed on the reactor, only the decontamination of tritium in the reactor containment is discussed in this paper.

To estimate the levels of contamination "powder" samples of the walls, floors and ceilings were taken using a special drill. The "powder" samples were immersed in distilled water for 24 hours, and then the tritium in the water was measured with a liquid scintillation counter to estimate the tritium concentration in the concrete. The minimum detectable limit of this method was 0.4 Bq per gram of concrete.

Contaminated concrete was removed with drills. Some concrete matrices were removed at a depth of several tens of centimeters.

Residual Radioactivity Criteria

Contaminated concrete was removed down to a level of residual radioactivity derived from the "limit for surface contamination" prescribed in an ordinance of the responsible Minister. The limit is 0.4 Bq per square centimeter for alpha-emitters and 4 Bq per square centimeter for other

radionuclides. The limit is one of the conditions for designating an area as a "radiation controlled area", where some restrictions for radiation protection purposes are to be implemented. This residual radioactivity criterion was chosen due to plans to release the reactor containment as a "radiation controlled area" temporarily, before it was reused as a containment for the new reactor. The residual radioactivity criteria were derived from the existing regulatory limit, without estimating any possible risk associated with this level.

It was confirmed that the residual radioactivity levels in the decontaminated concrete did not exceed the criteria. This was done by measuring tritium concentration in the samples of concrete powder using a method similar to that of the pre-decontamination survey.

The dose for an individual working for 2,000 hours per year in the reused reactor containment was estimated. With the level of residual tritium concentration measured after the removal of the reactor block, such exposure was estimated to result in an effective dose equivalent of about 3 micro-Sv per year due to inhalation of the contaminated air. The tritium concentration in the air of the reactor containment was assumed to be 70 micro-Bq per cubic centimeter.

Features of Health Risk

In this section, we identify some features of health risk which are encountered in reusing heavy-water cooled reactors. In the JRR-3 case, where a new reactor is to be installed and operated, the health risks of the reuse will result from both the operation of the new reactor and the residual radioactivity from the old. Although the risk due to the former will dominate, for the purpose of identifying general features of the health risk in reusing heavy-water cooled reactors, it is of value to consider the risk due to the residual radioactivity separately. The consideration will be informative for a case in which a contaminated reactor containment is reused for other than containing a reactor.

The residual radioactivity health risk peculiar to JRR-3 is from tritium. During a reactor operation, tritium penetrates concrete matrices in the floors, walls and ceilings. And during a period of reuse, the residual tritium escapes from the concrete to the air in the reactor containment. The dominant health risk of tritium is due to internal exposure from inhalation of contaminated air. External exposure of skin by submersion in the air will also be encountered. Skin can be directly exposed when it touches a contaminated surface. Moreover, if a ventilation system is installed in the reactor containment, environmental contamination outside of the containment caused by gaseous effluents containing tritium can also be a source of health risk.

One critical group, with respect to the health risk caused by a reuse of a heavy-water cooled reactor containment, will be the workers within the containment area. But under certain circumstances, the general public living around the reused facility can also be designated as a critical group.

Although the residual radioactivity criteria derived from the current regulation were implemented for JRR-3, alternative residual radioactivity criteria based on a risk assessment could be established. To establish residual radioactivity criteria in this way, a model would be needed for estimating the extent of risk. In the case of a heavy-water cooled reactor, the model should refer to the relation between residual tritium concentration in concrete matrices and airborne

tritium concentration in a room. In order to relate the concentration in concrete matrices to that in air, it is necessary to take a suitable average of the observed concentrations of tritium in the concrete matrices, because of "hot spots" of high contamination. Analyses of tritium behavior in a reactor containment, (i.e., its dynamics of diffusion and transport, effects of room temperature, humidity and ventilation) is indispensable in making a risk model. (One must analyse tritium behavior to calculate airborne tritium concentrations.)

The risk depends directly on the airborne tritium concentration in the reactor containment. A practical residual radioactivity criterion, therefore, would be a limit for airborne concentration or for the concentration in the concrete.

RESEARCH LABORATORY BUILDING NO. 1

Research Laboratory Building No. 1 was an assembly of chemical laboratory rooms where various radioisotopes were handled. It had a concrete framework and consisted of six stories, including two underground ones. In the first period of the decommissioning plan, the west half of the Building, which covered an area of 4,000 square meters with 70 rooms, was designated for decontamination and reuse. Unsealed radioisotopes were handled in the whole area of the third story and a part of the second story. Sealed radioisotopes were handled or X-ray generators were installed in the other rooms. The Building was used from 1959 to 1982 and therefore, most of the equipment and its function had lost its effectiveness. The Building needed to be repaired for maintaining safety and security. But it was judged to be impossible to reuse the Building as a radioisotope handling facility because most of the equipment and installations were too timeworn to handle radioisotopes. Therefore, the reuse plan for the Building was to remove all interior equipment and installations, to decontaminate its framework, and to reuse it as offices, meeting rooms or laboratories where radiation is not to be used. This was an example of releasing a building for unrestricted reuse. The decontamination lasted from December 1982 to March 1984.

Types and Levels of Residual Radioactivity

At the end of Research Laboratory Building No. 1's use, there remained various kinds of contaminated equipment: hot cells, fume hoods, desks, chemicals stockers, etc. Contaminated spots, generated by radioisotopes scattered during chemical handling, were distributed over the surfaces of walls, floors and ceilings. In the drainage system, contamination was distributed inside piping, and on floors and walls around leaky pipes.

Typical contaminating radionuclides were Ru-106, Cs-137, and isotopes of uranium and thorium. C-14's vapors contaminated wall and ceiling surfaces and the insides of some ventilation ducts. The maximum contamination level was measured at a spot on the floor. It was a Cs-137 contamination of 400 Bq per square centimeter. A room was entirely contaminated with C-14, where contamination was found on the floor, walls, ceilings and frames of the windows, with a maximum contamination level of 80 Bq per square centimeter. Only C-14 contaminated a whole room surface; contamination by other radionuclides were found only in spots or on fractions of room surfaces.

Decontamination

The decontamination process required for reuse in the Building included a survey of the levels of contamination, the decontamination itself, and measurement of the levels of residual radioactivity. Before the decontamination, in-situ surveys of levels and locations of contamination were conducted with gas-flow radiation counters with large detection windows. Alpha-emitters, i.e., isotopes of uranium and thorium, were also measured with these counters for their beta emissions. Radionuclides from the contaminated areas were identified with gamma-spectroscopy. For rooms where only one kind of radionuclide was handled, the nuclide could be identified from the history of the room, without consulting results of the spectroscopic analyses.

During the decontamination, all the interior equipment, piping, and ducts were removed. Loose or unfixed contamination on floor or wall surfaces was washed off. The contamination remaining after this washing treatment was removed with drills or scabbling equipment. Residual contamination was found to a depth of only a few millimeters, so it was easily removed.

Residual Radioactivity Criteria

Because of the planned unrestricted reuse of the Building, practically no residual radioactivity was allowed to remain. This requirement was attained and affirmed by an in-situ measurement of surface density of radioactivity throughout the entire decontaminated Building. The numerical criterion for the surface density of radioactivity was set equal to the minimum detectable limit of the radiation measurement instrument, i.e., 0.4 Bq per square centimeter for fixed beta contaminants. Finally, the Building was released from radiation control.

External and internal doses due to total surface contamination of 0.4 Bq per square centimeter of Cs-137 were estimated to be about 20 micro-Sv/y and 0.1 micro-Sv/y, respectively.

Features of Health Risk

In this section, we identify some features of the health risks resulting from residual radioactivity which would generally be encountered in the reuse of a large-scale radiochemical laboratory.

Observed patterns of residual radioactivity distribution in the Building fall into four categories: 1) extended contamination of inner surfaces of rooms (e.g., contamination with volatile radioactive materials, such as C-14), 2) hot spot contamination of surfaces, 3) localized surface contamination on and around piping or ducts for exhaust air or liquid effluents and 4) contamination on the equipment surfaces. Unfixed contamination will possibly spread over the building as well as to the outside environment through exhaust air or liquid effluents.

Health risks due to exposure to radiation from contamination can be encountered through the following pathways: 1) external exposure from the surface contamination, 2) internal exposure caused by inhalation of resuspended or volatilized radioactive materials from the surface contamination, and 3) external exposure through submersion in air containing resuspended or volatilized radioactive materials. Unfixed contamination can be easily decontaminated. Moreover, most of the fixed contamination remains at shallow depths in the building material and can easily

be removed. Thus, in the above-mentioned exposure pathways, those with volatile radionuclides will be most critical. Although the risks can expand to the environment outside of the building through radioactive liquid or gaseous effluents, the critical group will consist primarily of users of the Building following clean-up.

To establish a residual radioactivity criterion based on a risk assessment, a model which could express the relationship between surface contamination levels and exposure doses would be needed. This model should be applicable for various kinds of radionuclides. In addition, the model should provide a method of averaging various levels of spot or localized contamination, as well as contamination dispersed through ventilation or drainage systems. Modeling the behavior of volatile radioactive materials, such as C-14, would also be needed.

The risks depend mainly on the surface density of radionuclides on the floors, walls and ceilings. A practical form of a residual radioactivity criterion can therefore be expressed in terms of surface density.

CONCLUSION

Experiences in the decontamination and reuse of a heavy-water cooled reactor and a large-scale radiochemical laboratory were reviewed. The two reuse facilities discussed in this paper were quite different from one another

Following cleanup, JRR-3 was released for restricted use, but use of the Research Building No. 1 was unrestricted. It seems natural to set different levels of residual radioactivity criterion for the two types of reuse.

The main residual radionuclide in JRR-3, a heavy-water cooled reactor, was tritium. The Research Laboratory Building No. 1, on the other hand, a large-scale radiochemical laboratory, was contaminated with various kinds of radionuclides. Therefore, in order to set a residual radioactivity criterion nuclide-specific, radionuclides should be selected appropriately.

The two facilities also differed with regard to the health risk due to residual radioactivity. The health risk in a heavy-water cooled reactor was caused mainly by internal exposure resulting from the inhalation of tritium. In a large-scale radiochemical laboratory, the health risk was caused by both external and internal exposure to various radionuclides. To model these health risks, it is necessary to explain the relation between specific activity of a radionuclide and individual exposure dose for a heavy-water cooled reactor, and between surface density of radionuclides and individual exposure dose for a large-scale radiochemical laboratory.

If "operational" residual radioactivity criteria are based on some dose criteria, criteria for these two facilities should be different from each other in form.

It seemed appropriate to us that in the case of reusing buildings, an operational residual radioactivity criterion should be derived from a basic general criterion established by the regulatory authorities, in a case-by-case basis for each facility, according to its type of reuse, its main residual radionuclides, and its features of the health risk.

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Figure 1. Tritium Distribution inside Reactor Containment

Figure 2. Tritium Distribution in Concrete

*Applied Exposure Modeling for Residual Radioactivity and Release Criteria**

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ABSTRACT

The protection of public health and the environment from the release of materials with residual radioactivity for recycle or disposal as wastes without radioactive contents of concern presents a formidable challenge. Existing regulatory criteria are based on technical judgment concerning detectability and simple modeling. Recently, exposure modeling methodologies have been developed to provide a more consistent level of health protection. Release criteria derived from the application of exposure modeling methodologies share the same basic elements of analysis but are developed to serve a variety of purposes. Models for the support of regulations for all applications rely on conservative interpretations of generalized conditions while models developed to show compliance incorporate specific conditions not likely to be duplicated at other sites. Research models represent yet another type of modeling which strives to simulate the actual behavior of released material. In spite of these differing purposes, exposure modeling permits the application of sound and reasoned principles of radiation protection to the release of materials with residual levels of radioactivity. Examples of the similarities and differences of these models are presented and an application to the disposal of materials with residual levels of uranium contamination is discussed.

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INTRODUCTION

Modeling studies of the fate of radioactive materials released to the environment have been performed over the past forty years. Within the past decade, they have been applied to the establishment of regulatory criteria and standards. Models are a useful means for examining the consequences of the release of radioactive materials because they provide a consistent presentation of the significant transport mechanisms and exposure pathways affecting human health and the environment. ICRP [1,2] has provided the basis for evaluating doses on a consistent basis with the application of constant risk from exposure to radiation. This has allowed models to examine scenarios for the exposure of workers and members of the public and compare the relative doses providing a technically defensible basis for establishing regulatory standards. Maximum individual doses and collective doses to larger groups of people can be addressed by models on an equivalent basis to provide consistent levels of protection appropriate to the waste management practice being considered. Exposure models include doses received from external exposure, inhalation, and ingestion pathways and are typically tailored to the specific management practice and radioactive materials being evaluated. The application of exposure modeling to residual radioactivity and recycle provides a means for addressing many of the unresolved issues in current regulations, such as the criteria for the release of materials with bulk activity compared to those with surface activity. With a reasonable representation of the source terms, bulk and surface activity can be addressed separately and the resulting consequences evaluated on the consistent basis of potential doses to a member of the public.

Exposure models differ in the detail and depth of analysis, but all exposure models share the same basic conceptual elements. Many of these elements require the use of approximations or assumptions that limit the validity of the model for some applications. For the application of exposure models to residual radioactivity and release criteria, the identification of a dose limit is an important consideration. Without the identification of a dose limit, model results are difficult to evaluate and compare, which leads to results which are likely to be suggestive rather than conclusive. Another common element in exposure modeling is the definition of the source of radiation. Often this aspect of modeling is the most important and difficult component of the analysis. Models typically resort to some level of approximate analysis with respect to the source term that is subsequently propagated throughout the analysis. Assumptions invoked in defining the source term are typically the best indicator of the range of applications that an exposure model can be used and remain technically defensible. Having an established source term, models require the identification of exposure scenarios likely to occur as a result of the waste management practice under consideration. These scenarios are identified with the intent of encompassing potential exposures to the public and the environment in the future and are formulated with a conservative interpretation of possible events. With the establishment of the exposure scenario, the source is transported through environmental media to the point of human exposure by a modeled representation of the facility and site performance. At the point of human exposure, the concentrations of contamination are related to doses using the exposure scenarios identified. Model results alone rarely provide a definitive answer of the potential dose as compared to the prescribed dose limit because of the assumptions and approximations incorporated into the model. Consequently, interpretation of the results is necessary to support any decision resulting from the application of exposure modeling.

While exposure models all share the same basic conceptual elements, the variety of applications and different concerns addressed have resulted in a large number of models to consider in model selection. In spite of the extensive list of available models for performing exposure analysis, the existing models are of three basic types. These types can be regarded as generalized or generic models, simulation or research models, and compliance or site-specific models. Each model type has an important role in understanding the consequences of management decisions regarding radioactive materials. Generic models provide a general interpretation of potential exposures from a waste management practice and are useful in the support of rulemaking, such as the modeling used to support 10 CFR 61 [3]. Research models provide scientific simulation of the fate and behavior of radioactivity using best available scientific analysis of important components of exposure models and are useful for advancing the state-of-the-art in modeling potential exposures. Site-specific models provide conservative estimates of facility and site performance for demonstrating compliance with regulatory requirements for specific applications. This type of modeling is useful for demonstrating protection of public health and the environment when waste, facility and site characteristics are important considerations in determining the acceptability of the waste management practice. These site-specific factors frequently are critical to making decisions that are both appropriate and defensible.

Modeling potential exposure from a waste management practice provides forecasts of future consequences of the release of radioactive materials. In the evaluation of the consequences of the release of materials with residual levels of radioactivity, the potential uses of these materials following release is limitless. Once released, control is difficult to impossible to restore. Modeling can not and should not be expected to predict the future use of materials released for recycle. Since materials that are recycled will ultimately be disposed of as ordinary waste and any waste may be salvaged for reuse, a conservative approach to exposure modeling that is defensible is to consider all materials with residual levels of radioactivity as wastes that are disposed of and subject to salvage in the future. By considering future salvage, exposure modeling needs to address the public adjacent to the disposal facility as well as an inadvertent intruder who comes in contact with the waste.

APPLICATION OF EXPOSURE MODELING TO BRC WASTE DISPOSAL

An approach to exposure modeling using generic models for BRC waste disposal has been prepared by EPA in support of the proposed 40 CFR 193 rulemaking process. The exposure model used to support the proposed rule evaluates doses in the context of risk to provide support to the selected dose limit for BRC waste disposal. A site-specific exposure model for a site with a defined waste stream is in contrast to a generalized exposure model. A site-specific model applied to the BRC disposal of uranium contaminated wastes from the Y-12 Plant in Oak Ridge illustrates the differences between generic and site-specific exposure models [4].

As a first step in defining a site-specific model, an annual dose limit of 4 mrem effective dose equivalent from all pathways was selected, which is consistent with the proposed BRC limit being considered by EPA. Since the wastes to be evaluated include only depleted uranium (0.02% U-235), modeling needs to focus on the likely pathways of exposure associated with long half-life materials where the predominant doses result from ingestion and inhalation of

contaminated material. The disposal concept is to have a dedicated facility within the DOE Reservation in Oak Ridge which limits inadvertent intrusion into the waste until governmental control of the Oak Ridge Reservation is lost. Exposure modeling requires the evaluation of doses by considering concentrations of contamination; however, uranium is managed within the Y-12 Plant on the basis of activity. The management of uranium on an activity basis is important for uranium accountability requirements but necessitates the conversion of uranium activity to concentration by the use of the waste density. This conversion is not easily made since waste materials are likely to have a broad range of densities depending on the generating process. These site-specific issues and the unusual site characteristics of the Oak Ridge Reservation are not typically included in generic models. The use of generic exposure models for this application would necessitate extensive modifications to existing generic models. A more comprehensive technique for ensuring that important site-specific conditions are presented as part of the exposure model is to utilize site and facility data to define the scope and detail of the exposure model. By focusing on the known data and associated issues, site-specific analysis can become an extremely useful tool in waste management for examining alternatives and developing management practices that are acceptable to regulatory requirements and protective of public health and the environment.

Site-specific exposure modeling is typically founded on the data that are available to examine potential exposures; however, numerous assumptions are typically required to provide a complete picture of the waste management practice. Assumptions are chosen to be conservative but not all assumptions are well supported and can only be regarded as best estimates. For the application to Y-12, these assumptions are described for offsite and intruder exposure scenarios, site performance, waste form and the disposal facility. The details of the assumptions are shown in Table 1. While many assumptions are invoked the use of data is maximized. Important site-specific data are incorporated into the transport calculations that are determined from site characterization investigations of the proposed disposal site. The combination of the assumptions and data are used to define the limiting concentration of uranium in a dumpster unit that contains uranium contaminated materials and meets the prescribed dose limit. The resulting dumpster loadings and activity limits are presented in Table 2.

DISCUSSION

The preceding exposure modeling methodology illustrates the differences in detail and approach between site-specific and generic modeling. The most striking difference is the focused orientation of the site-specific analysis in contrast to the exhaustive analyses common to generic models. Emphasis is placed in site-specific modeling on those factors that are significant to the protection of public health and the environment to guide acceptable management decisions. In humid environments, where the dominant pathway for the transport of contamination is by surface water and groundwater, examination of the potential for atmospheric exposure is tangential to selecting acceptable management practices. Similarly, the consideration of decay and direct exposure from the wastes is not a major consideration for uranium contaminated wastes. Uranium contaminated wastes provide an excellent example of the usefulness of site-specific analyses because uranium is a ubiquitous element that needs to be addressed as a contaminant in the context of the increment over natural background. Additionally, the activity of uranium contaminated materials can vary widely depending on the isotopic composition of the uranium source material. Failing to consider the specific differences

between the various forms of uranium can lead to widely varying results. The unique features of uranium need to be considered separately along with the unique institutional and site characteristics associated with uranium contaminated materials that could be considered for management as BRC wastes.

Exposure modeling for site-specific applications can be anticipated to yield results that are representative of the materials to be considered for release with residual levels of radioactivity. Results from site-specific analyses can be expected to yield results that differ from generalized models whenever material properties or site characteristics are unusual. Since one or the other of these conditions is commonly encountered, generic modeling results can be misleading unless the results are subjected to careful interpretation. Likewise, results from site-specific modeling need to be carefully considered against results from generic models to ensure site-specific models incorporate all of the significant pathways of exposure. Research models have an important role in determining the appropriateness and correctness of the simplified model components typically invoked in generic or site-specific models. Consequently, each type of exposure model has an important role in providing protection to public health and safety, and no one type of model should be ignored or excluded when considering an approach to modeling a specific disposal system.

CONCLUSIONS

Exposure modeling provides a basis for the application of sound and reasoned principles in radiation protection to the release of materials with residual levels of radioactivity. Exposure modeling supports criteria for the release of materials that are technically defensible and consistent while having the flexibility of examining specific or unique applications. As a result, exposure modeling supports rulemaking, the determination of compliance with existing rules and the improved understanding of the consequences of the release of materials with residual levels of radioactivity. By the use of generic exposure models, criteria can be established that define overall performance objectives and specific exemption levels. Exceptions from specific exemption levels for unique materials or sites can be accommodated by the application of site-specific exposure modeling.

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Table 1
Assumptions Applied to BRC Disposal of Depleted Uranium

Offsite Exposure Scenario
- Contaminated groundwater consumption
- Point of consumption at facility boundary
- Exposure occurs at highest concentration

Intruder Exposure Scenario
- Intrusion occurs 50 years after facility closure
- Foodstuffs consumed that are grown in contaminated soil
- Contaminated groundwater consumption
- Contaminated milk and meat are consumed
- External exposure and inhalation

Site Performance
- Dilution of leachate occurs in unsaturated zone by infiltrating precipitation
- Consumption of contaminated water occurs at the top of the saturated zone without aquifer dilution
- 50% of site surface area is dedicated to disposal
- Dilution of leachate determined by application of FEMWASTE code

Waste Form and Disposal Facility
- 50% of waste is soluble
- Release of soluble portion occurs uniformly over a 10-year period
- 50% of the leachate generated is collected by the leachate collection system
- Waste is diluted by a factor of 4 within the disposal unit by uncontaminated materials
- Waste resembles soil at the time of intrusion
- Waste and uncontaminated soil are uniformly mixed across the disposal site area upon intrusion
- Average concentrations in disposal units are the concentrations in each and every dumpster disposed of at the disposal facility
- FEMWASTE assumptions that include:
  - Leachate concentrations controlled by solubility limits
  - Kd determined by lowest experimental equilibrium value

Table 2
BRC Limits for Depleted Uranium Calculated by Exposure Modeling

Dumpster Size (Cubic Yards)	Activity (uCi)	Mass (g)
6	200	400
10	300	700
12	400	900

DOE Guidelines and Modeling in Determination of Soil Cleanup Guidelines

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ABSTRACT

This presentation will summarize the current guidelines used by the Department of Energy for their remedial actions under the Office of Nuclear Energy's Formerly Utilized Sites Remedial Action Program and Surplus Facilities Management Program. It will discuss soil criteria and surface contamination limits used for buildings and equipment. A brief description of the computer code, RESRAD, which is used to develop soil criteria and estimate doses from residual radioactivity will also be provided.

INTRODUCTION

The Department of Energy currently implements remedial actions and decontamination projects at its Formerly Utilized Sites Remedial Action Program (FUSRAP) sites and its Surplus Facilities Management Program (SFMP) sites using the "Department of Energy Guidelines for Residual Radioactive Material at FUSRAP and SFMP Sites", Revision 2, March 1987. The Guidelines are supported by a computer code, RESRAD, which is used to estimate potential doses from residual radioactive material and support the development of soil cleanup criteria. This presentation will provide a brief history of the development of the guidelines, some general discussion of the guidelines and a general overview of the RESRAD code.

BACKGROUND

In the late 1970's and early 1980's the Department of Energy began the survey and ultimate decontamination of several facilities that had been used by the Departments predecessor agencies in the development of nuclear energy. Criteria and standards for these sites were proposed on a site-by-site basis by the field offices responsible for the specific remedial action. The criteria and approaches varied from site to site. In reviewing the criteria DOE headquarters

realized that general guidelines were needed and at their request the national laboratories made several attempts to generate generic criteria. In 1983, DOE consolidated these criteria and procedures in order to identify generic soil criteria for all remedial actions.

The DOE headquarters established a working group comprised of representatives from concerned program offices; the Office of Environment, Safety, and Health; DOE Operations Offices; and the involved national laboratories. The working group activities were also coordinated with the U.S. Environmental Protection Agency and the U.S. Nuclear Regulatory Commission. These representatives attended several meetings of the working group in an advisory capacity.

The initial charter of the working group was to develop an acceptable set of generic soil criteria. However, the recommendations from the first working group meeting resulted in an action which expanded the charter to cover all aspects of decontamination. Because potential doses from residual radioactivity are dependent on many site specific factors, the working group also recommended that a generic procedure for developing soil limits be developed rather than generic concentration limits. This recommendation resulted in the development of the RESRAD computer code which supports the development of soil criteria. The following is a summary of the working group's major recommendations:

The guidelines should be consistent with other available standards where they are appropriate including the EPA standards for Uranium Mill Tailings Remedial Action (UMTRA) (40 CFR 192); NRC surface contamination limits (Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source or Special Nuclear Material, July 1982); and DOE Orders.

It was recommended that DOE adopt the standards, guidance and models in publications 26 and 30 of the International Commission on Radiological Protection (ICRP). This includes the use of the concept of effective dose equivalent rather than critical organ dose (ICRP publication 2) for defining dose limits and that a 100 mrem/year effective dose equivalent be adopted for longterm exposures instead of the 500 mrem/year whole body limit then in effect.

Soil criteria for radionuclides other than those covered under 40 CFR 192 (radium and thorium) should be derived for each site to ensure compliance with the dose limit. These criteria should be based on a conservatively assumed plausible-use (realistic) scenarios.

The as low as reasonably achievable (ALARA) process should be incorporated into the guidelines for all phases of a remedial action.

No attempt should be made by DOE to define "de minimis" or "below regulatory concern" levels.

GUIDELINES

The guidelines were developed in accordance with these recommendations and were first issued to the field for implementation in February of 1985. The current guidelines are revision 2, dated March 1987 (they are included as DOE-wide guidelines in DOE Order 5400.5, "Radiation Protection of the Public and Environment"). The revisions were generally procedural in nature and were directed toward resolving implementation problems experience with the earlier versions.

The guidelines include a primary dose limit of 100 mrem/year for remedial actions and a requirement to bring potential doses as far below that limit as is reasonably achievable (the ALARA process). It also adopted as secondary limits, the 40 CFR 192 concentration limits for radium and thorium in soil as well as the radon standards for buildings. The Nuclear Regulatory Commission (NRC) surface contamination limits ("Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source or Special Nuclear Material", NRC, July 1982) were included as guidelines to be used for the release of usable buildings and equipment.

Soil criteria for radionuclides other than radium and thorium are derived and are applied over each 100 square meter area of a site. Hot spot criteria are also included to ensure no adverse impacts will result due to averaging the residual radioactive material over 100 square meters. Guidance is provided for interim and longterm management of remedial action residues and wastes.

Procedures for supplemental limits or exceptions to generic or derived limits are also provided in the guidelines to address those situations where special site-specific circumstances indicate they are required. In general, the use of exceptions and supplemental limits is discouraged. Exceptions are used in those cases where remediation cannot effectively reduce residual material to levels that will allow use without radiological restrictions. An exception may be granted when the above condition is demonstrated and when it is shown that partial remedial measures coupled with controls on use of the site will adequately protect the public. Supplemental limits are applied to a portion of a site or facility for which the derived or generic limits are not appropriate and because of the nature of the area either more stringent or less stringent limits are necessary. However, supplemental limits do not consider use restrictions in their development. The steps and requirements for justification of either exceptions or supplemental limits are described in detail in the guidelines.

MANUAL AND RESRAD COMPUTER CODE

As noted above the guidelines are supported by a separate but integral implementation manual (DOE/CH/8901, "A Manual for Implementing Residual Radioactive Material Guidelines") and computer code (RESRAD). The manual describes the pathway methodology to be used in deriving soil criteria and the associated dose and transport factors necessary for their development. It provides guidance on the application of the ALARA process and on the use of hot spot criteria. The manual also describes the computer code RESRAD and its use.

RESRAD can be used to calculate site-specific soil guidelines on the basis of any selected dose limit. It may also be used to calculate doses to a hypothetical on-site resident or worker (maximally exposed individual member of a critical population group).

Dose calculations are done for the external radiation exposure pathway and the internal exposure pathways including ingestion of drinking water (surface or groundwater), vegetation, meat, milk, and fish, and inhalation. The program contains vegetable/soil transfer factors, food transfer factors, radionuclide distribution coefficients, dose conversion factors and decay and ingrowth data. Input parameters include:

Physical parameters such as size, thickness, density, and porosity of contamination, the cover and subsurface layers.

Hydrological parameters including conductivity, gradient, and water table depth.

Geochemical parameters such as distribution coefficients.

Meteorological parameters including precipitation, evapotranspiration, erosion, runoff and mass loading.

Usage and consumption parameters for the scenarios.

All of the input parameters can be site-specific and entered by the user. Under circumstances where information on the site is limited, default values can be used. However, the dose conversion factors, and the uptake and transfer factors are standard and are not readily changed by the user.

The current version of RESRAD and the associated methodology does not address the indoor radon pathway. At present the DOE Guidelines have adopted the 40 CFR 192 radium limits and as a result, this pathway has not yet been included. The code does not compute collective dose to offsite individuals. If the site under review is large enough that such evaluations are warranted other models are used for offsite dispersion.

The code runs on IBM Compatible PC and PS/2 computers with a hard disk drive and 400K of memory. A math coprocessor and mouse are recommended but not required. The program is menu driven and very user friendly. Internal interactive help files are available.

SUMMARY

Use of the DOE guidelines (100 mrem/year plus ALARA) with RESRAD and the associated methodology described in the manual to derive soil limits for DOE remedial actions has been successfully applied in the field. Generally, during planning, an authorized concentration limit for cleanup is selected at a level that would result in doses, under current use or realistic use scenarios, that are well below the 100 mrem/year dose limit. The ALARA process is also extended to field operations which ensures that the user of the site or area will not receive doses anywhere near the 100 mrem/year limit.

While this code, the associated methodology and the supporting manual were developed to support the DOE Guidelines, the methodology and code can be a useful tool for determining potential doses to individuals from soil contamination and in evaluating various remedial action options. The code and methodology offers a consistent approach with considerable site-specific

flexibility. Once the site specific data has been entered into the data fields, the program computes potential doses and soil guidelines (based on a specified dose) quickly and, hence, can be used as a tool to optimize cleanup options or even to support certain ALARA process determinations.

Session IV

Desirable Characteristics for Criteria

International Similarities and Differences in Regulating Nonradiation Hazards

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ABSTRACT

Assessment of the risk of cancer and other health effects, and of injury or other deleterious consequences of technological hazards, involves intricate interactions between science and policy. These interactions are best understood in terms of (1) a gradual overlay or replacement of traditional administrative approaches to regulation with "scientific" principles and methods, and (2) the emergence of different styles or approaches to regulation in different countries.

All regulatory styles involve some mix of consultation, reliance on expertise, self-regulation, and consideration of political context. But variation in their respective contribution in different countries leads to characterization of "typical" regulatory styles: evidential (United States); consensual (Sweden, United Kingdom); authoritative (France, Japan); corporatist (The Netherlands, West Germany). The persistence of these political and administrative traditions suggests that they will hold for other areas of regulatory action.

INTRODUCTION

Many people, when travelling in foreign countries, observe first-hand that things work differently overseas. This was brought home to me rather forcefully on my first visit to Britain. I had been out of college only a couple of years, and was to visit the London office of the contract research firm for which I worked. When I arrived at my hotel, a message from the head of the local office was waiting. It said, "Please ring before you call." I was utterly baffled. Fortunately, I had been a student who plagued his professors with questions. I asked the concierge at the hotel what the message might mean. Of course he was able to provide a translation: I should telephone the office before visiting.

The Japanese visitors at this meeting may have had no trouble with such a message. But for Americans, both "ring" and "call" are used most commonly in reference to the telephone, which obviously is not the case in Britain. Such differences in American and British English are common. It has been said that the United States and the United Kingdom are two countries separated by a common language.

Confusion based on different use of the same terms is not limited to everyday experience. For years I have studied the use of scientific evidence and judgment in regulatory decisions in different countries. I have observed on several occasions the frustration of European researchers as they attempted to gain wider application of some of the concepts of risk assessment and cost-benefit analysis. I believe they often used the same concepts to describe quite different procedures because they were thinking in terms of the processes that occurred at home in their own countries. Since those processes differed, they had slightly different, sometimes quite different, meanings for the concepts.

Today I will address similarities and differences in the regulation of nonradiation hazards in different countries. The meeting organizers originally asked me to talk about similarities and differences in the clean-up of residues of radioactive materials at storage and disposal sites. However, despite considerable effort, I have been unable to locate sufficient published materials on that subject. So I must rely instead on topics I have studied and can discuss with some degree of authority. I hope that the participants will be able to extend my observations to the topic of the workshop.

DIFFERENT RELIANCE ON SCIENTIFIC EVIDENCE AND ADVICE

On August 8, 1987, a resident doctor of the hospital at Osaka University received a liver transplant at the Cromwell Hospital in London.[1] Since no definite policy on organ transplants existed in Japan, the transplant had been arranged overseas. There was no study comparing the risks and benefits of liver transplants and other more conventional therapies in Japan. This is in sharp contrast to the situation in the United States, where statistics on success rates and other indices of medical performance are commonly available for separate hospitals and sometimes individual physicians.

This example highlights an important difference between the typical approaches to pubic health policy in the two countries. Japan seems to rely on guaranteed safety based on an established, formal set of relationships among organizations and individuals rather than on explicit information about the specific situation as is typical in the United States. Furthermore, the general preferences in the two countries apply in a variety of settings. We can anticipate that they will influence the development of the radiation clean-up criteria under discussion at this workshop as well.

To set the stage for a more detailed look at the relevant influences on regulatory approaches, let me describe what has been called the "emerging scientific state."

THE EMERGING SCIENTIFIC STATE

Since the 1960s, it has been common to consider interactions between science and public policy in terms of both "science for policy" and "policy for science."[2] The

concept of science for policy refers to the fact that modern governments depend on science to perform, or at least provide input for, many of their traditional functions. Policy for science derives from the emergence of science as a national resource. It is the first of the two, science for policy, that concerns us here.

In describing what they call the "scientific state," Schmandt and Katz [3] identify three principal characteristics of the interactions between science and policy: (1) science affects the public policy process in distinctive ways that can be identified as stemming uniquely from the realm of science; (2) the resulting changes gradually replace, modify, or supersede, as the case may be, older agenda, organizations, and procedures of policy making and administration; and (3) science induces change in all industrial countries, but the resulting social environments differ, depending on national circumstance, traditions, and goals. Schmandt and Katz point out that science enters the policy process in three distinctive ways: as product, as evidence, and as method.

The search for socially acceptable policy responses to complex and uncertain hazards often pits new options against more familiar legal and political options. Science is often seen as providing these new policy options. In other words, science produces policy products for consideration.

But in another sense, or perhaps in other instances, science provides information about all the options under consideration. The interpretation of scientific evidence concerning the likely consequences of alternative options is a central facet of contemporary industrial states. The political and administrative systems vary from country to country, however, and it should be no surprise that the way these interpretations affect policy decisions also vary.

Finally, many activities involved in policy making are based on explicit and logical analysis. From science comes formal methods for analysis of quantitative data, gathered and interpreted according to accepted method. From the policy environment comes the requirement for timely action and the acceptance of uncertainty, incomplete information, and compromise.

Each way that science enters the policy process--as product, as evidence, and as method--interacts with the traditions found in various countries.

COMMON CHARACTERISTICS OF REGULATION

Although there are differences in the political and administrative traditions, the basic issue--the choice among complex and uncertain options--is similar everywhere. Four major characteristics describing regulatory approaches used in all countries have been identified: consultation, dependence on expertise, self-regulation and self-policing, and political considerations.[4]

Some form of dialogue or consultation among the affected parties is a feature of all regulatory approaches, although there are notable differences in what is discussed, when, and with whom. Consultation is often an unavoidable part of the political and

administrative system, but even when not legally required it is often used to help reach agreement about the proposals that are most likely to succeed.

Experts with special understanding and knowledge provide vital input to all regulatory processes. The nature of interactions between experts and public officials vary among countries, however, and attitudes about expertise, consultation, and participation are changing the nature of the expert's role.

Some degree of self-regulation and self-policing is a part of every regulatory regime. But some countries rely on self-regulation to a much greater degree than others, making it a virtual cornerstone of their whole approach.

In the past, regulatory decisions were commonly thought of as primarily technical issues. Now, in contrast, political and social factors often appear to outweigh technical concerns. On occasion, options are chosen more for their political palatability than their technical feasibility.

Let me now turn to the way approaches to regulation vary.

TYPICAL STYLES OF REGULATION

Four main approaches to regulation have been identified: evidential, consensual, authoritative, and corporatist.[5] A particular style is usually dominant in a given country, although different styles may be adopted by some agencies or organizations within the same country. There does appear to be a tendency toward convergence among countries, with less marked differences than was the case a few decades ago. This would be consistent with the notion that the "scientific state" is gradually replacing more traditional administrative procedures throughout the industrialized world.

THE EVIDENTIAL APPROACH

In the evidential* model, for which the United States is the prime example, regulatory approaches result in the involved parties becoming adversaries, often arguing the facts and interpretations of them in legal battles. Because of these conflicts, regulations are increasingly formulated in terms of precise targets, standards, and procedures. There is also separation of powers, and dispersion of authority among units and levels of government. There tends to be complex structures of interest groups inside and outside government, and the configuration of those coalitions shift with issues rather than party politics.

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* This category is commonly labeled "adversarial" in the literature. I believe, however, that "evidential" more accurately describes the key factors that underlie its differences with other approaches.

THE CONSENSUAL APPROACH

A regulatory approach aimed at achieving consensus may confine its consultation to an elite group of civil servants, experts, and influential politicians and industrialists, as in the United Kingdom. Or it may extend involvement more broadly, as in Sweden. However, the elite consensual approach, common in several European countries, differs from the evidential approach in the United States in several ways. Explicit specifications and formal rules are avoided, allowing government considerable discretion when addressing specific circumstances. Procedures tend to be more intrusive--in order to deal with each case individually, it is necessary to have detailed information about it. Dealing with each case in detail requires considerable confidentiality, especially with regard to proprietary commercial data. Both intrusiveness and confidentiality encourage self-regulation with little legal prosecution but considerable, and generally effective, moral suasion. The consensual approach often results in development of an "elite club" of those involved, which treats those not involved with disdain or even contempt. This makes it doubly difficult for outsiders to influence the development of consensus politics.

THE AUTHORITATIVE APPROACH

Regulation by centralized authority is most often found in countries like France or Japan which have a strong central government and a weak legislature, and where regional or local government is largely limited to executing decisions issued by national bodies. In this approach, regulators have considerable leeway in setting standards and enforcing compliance. Consultation with relevant parties tend to be infrequent, although those that take place tend to be crucial to the outcome. Regulations are usually imposed and enforced with limited scope for legal redress unless officials can be shown to have acted arbitrarily or contrary to statutory procedures. The implementation of regulations generally leave room for negotiations recognizing the unique attributes of the specific situation.

THE CORPORATIST APPROACH

The corporatist regulatory style, most often associated with the Netherlands and West Germany, combines aspects of the analysis found in the evidential approach with the consultation found in consensual models. The general framework of regulations, and sometimes the specific content, are established in groups where individuals "represent" the relevant economic or social interest groups. There are many consultative bodies and advisory mechanisms. These groups tend to be construed in ways that correspond to the configuration of relevant interests. Evidence is used more to determine the boundaries of choice rather than determine the preferred outcome. Debate may be open to anyone, but decisions are generally in closed groups so as to maintain the established balance of interests. Corporatist approaches tend to restrict the influence of smaller or newer groups.

PRACTICAL IMPLICATIONS OF DIFFERENT APPROACHES

Let me now speculate about the implications of these differences in approach for the development of criteria for radioactive materials in Japan and the United States. If

control of air pollution can serve as an example, we may expect Japan to formulate general goals for clean-up and to specify levels of contamination as "guidelines." Although this process would involve considerable analysis and consultation with experts, the relationship of the two might not be extensively documented. The determination of the exact level of permissible residual contamination would be determined on a case-by-case basis, subject to extensive negotiation. The United States, in contrast, is more likely to develop explicit levels of permissible contamination for general application in a lengthy rule-making procedure subsequently defended in court. The rule-making decision would likely refer to extensive evidence and support analysis. Extensive records would likely document the handling of each site.

CONCLUSIONS

In this short talk, I have described several aspects of the similarities and differences among the approaches to regulating nonradiation hazards in different countries. I also briefly suggested how these observations might apply to the development of criteria for clean-up of radioactive materials. My comments have probably raised more questions than provided answers. I would be happy to try to answer any questions you might have.

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What are the Basic Requirements that Cleanup Standards Should Satisfy?

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It is essential that we, as a society, be clear about what we expect to achieve through standards for cleanup. Fortunately, in radiation protection, as opposed to protection against other environmental threats, we have been thinking about the basics for a long time. As a result, our objectives are well developed.

Most of these objectives were laid out almost forty years ago, in the late 1950's, and have changed only slightly since then. There are three; they were first set forth as guidance for regulation of radiation exposure by Federal agencies, by the Federal Radiation Council, in 1960 (Figure 1):

1. "There should not be any man-made radiation exposure without the expectation of benefit resulting from such exposure."

2. "Every effort should be made to [keep] radiation doses as far below [the recommended limits] as practicable."

3. "[Recommended limits] for the general population are:
   0.5 rem per year (whole body)
   5 rem in 30 years (gonads)"

Figure 1. U.S. Federal Radiation Council (1960)

These three objectives have been characterized, by the International Commission on Radiological Protection (ICRP), as the principles of justification, optimization, and limitation; we

will use that convenient terminology here. Justification is a judgment that is applied when a practice involving the use of radiation is introduced. In the 1960 Federal guidance, it was stated as a requirement that there should not be any radiation exposure without the expectation of a net benefit. This requires a broad judgment. Often, it is a political one. It is generally not made by radiation protection professionals, although they usually have the opportunity for input.

The second objective, optimization of protection, is more commonly known in the United States as the ALARA ("as low as reasonably achievable") principle. It has been with us since the early 1950's, and flows from the assumption that there is a linear relationship between radiation dose and the risk of deleterious effects on health. This assumption leads to the conclusion that, since any dose carries some risk, dose should be maintained at the lowest level that is economically reasonable.

The third objective is limitation of risk to the individual. In the 1960 Federal radiation protection guidance it is expressed in terms of dual limits on dose to members of the general population, one for exposure of the whole body and one for genetic exposure. It must be satisfied without regard to the economic cost. This third requirement is needed to supplement the first two because they assure only that the effects of radiation on the population as a whole is reduced to reasonable levels. They do not assure that the distribution of these effects is such that no individual receives an unreasonably large share of the risk from exposure to a particular source.

These three principles were set forth in essentially the same form by the ICRP in their Publication No.2 in 1959, the year before the 1960 Federal guidance was issued. We have noted that there were dual limits for the individual. These disappeared in 1976, when the ICRP revised its recommendations (ICRP Publication No.26), in the next major statement of basic principles. This is shown in Figure 2.

1. No practice shall be adopted unless its introduction leads to a positive net benefit.

2. All exposure shall be kept as low as reasonably achievable, economic and social factors being taken into account.

3. The dose equivalent to individuals shall not exceed the limits [0.5 rem in any year, and, in the case of chronic exposure, 0.1 rem per year.]

Figure 2. International Commission on Radiation Protection (1977) (as interpreted by the "Paris Statement" of 1985)

The three principles remain. However, one important distinction was introduced in the application of limitation. The ICRP now makes a distinction between limits for short-term and for long-term exposure. There is one limit for chronic exposure, say, from a planned release from a fixed facility, and a second limit for non-recurring short-term exposure, e.g. from a contaminated site prior to cleanup. This distinction evolved gradually from the 1976 ICRP recommendations, and it was not stated explicitly until 1985.

Notice that the phrase "as low as practicable" in the former FRC Guidance has now become "as low as reasonably achievable, economic and social factors being taken into account," a much more realistic statement of this requirement.

As some of you know, the Environmental Protection Agency (EPA) inherited the functions of the former Federal Radiation Council when EPA was formed in 1970, and in 1987 we revised the Federal Radiation Protection Guidance for workers. These new recommendations replaced that portion of the 1960 Federal Guidance that applied to workers. It was developed cooperatively by all of the affected Federal agencies.

We have also recently convened a new Federal interagency group to develop guidance on residual radioactivity. As a first task this group is developing recommendations to update the remaining portion of the 1960 Federal guidance, which applies to the general public. That work is now near completion. Figure 3 shows excerpts from the current draft. Although I can guarantee that the words will change before it is published for public comment, the basic ideas should remain the same.

"For those sources of exposure covered:

1. "There should be no exposure of members of the general public . . . unless it is justified by the expectation of an overall benefit . . .

2. "A sustained effort should be made to ensure that doses to individuals and in populations are maintained as low as reasonably achievable."*

3. "The combined radiation doses incurred in any year from all sources of exposure . . . should not normally exceed a limiting value of 0.1 rem . . .

* "[This] includes consideration of economic and social factors, and applies to radiation exposure that may occur now and in the foreseeable future."

Figure 3.
Draft Revision of Federal Radiation Protection Guidance (1989)

As you can see, the basic objectives have not changed dramatically in the thirty years since 1960. There is, however, one significant change that is not embodied in previous statements of basic principles. That is, a distinction is now made between limits on dose from individual sources of exposure and limits that apply to the sum of doses to an individual from all sources. This distinction has been evolving in the ICRP, as well as in regulatory practice in the United States. The ICRP now places primary importance on limits that apply to individual sources, and much less on limits for total dose to individuals. Limits for individuals apply to all of the various sources of exposure that an individual may be subjected to, and, of course, is achieved -- in a regulatory fashion -- by applying limits to individual sources. A consequence of this, of course, is that no individual source can take up all of the limit on allowable dose to an individual.

This was expressed well in a recent publication of the International Atomic Energy Agency (IAEA) (Figure 4). What they said is perhaps obvious -- the problem is to account for the presence of other sources, the continued operation of these other sources in the future, and the introduction of new sources. For this reason, they introduced the idea of a source-related limit that is lower than the dose limit -- a source upper bound.

> "Since [the primary dose] limit is individual-related . . ., account must be taken of the presence of other sources, the continued operation of these sources in the future, and the eventual introduction of new sources. For this reason a source-related limit, lower than the dose limit and called a source upper bound, must be set . . . as the boundary condition for optimizing." (IAEA Safety Series 77, 1986)
>
> "Authorized limits for sources will normally be a fraction of the limiting value for individual dose from all sources combined." (Draft revision of Federal guidance, 1989)

Figure 4. Source-related limits

Notice that this is not the limit for sources; it is just the upper bound for source-specific limits and actual limits will generally be lower, because they also must take account of optimization of control. In the draft revision of Federal guidance for the general public (Figure 4) we have addressed these problems through simple recognition that authorized limits for sources will normally be a fraction of the limits for dose from all sources combined.

The distinction between individual and source-related limits has been recognized in the U.S. for some time. It is reflected by a series of source-related standards established during the last decade and a half -- usually in response to public opinion expressed in the form of statutory mandates from Congress. The standards established under the Clean Air Act (40 CFR Part 61) for a host of types of sources of radionuclide emissions is the most recent example. Previous

examples include the EPA's standards for the nuclear fuel cycle (40 CFR Part 190), high level waste (40 CFR Part 191), and uranium mill tailings (40 CFR Part 192).

We have not had to give serious attention to the idea of a source upper bound in the U.S. because most of the standards we have set have been based, at least in part, on the application of optimization, and have resulted in values that were low compared to individual-related limits. That is, the individual limit in this country, which is still five hundred millirems per year, has never been in danger of being exceeded. This would still be the case with an individual limit of 100 millirems per year, as currently recommended by the ICRP.

For residual radioactivity this may no longer be true, because it is highly unlikely that optimization considerations are going to be controlling. That is, individual risk is going to be the most important consideration for residual radioactivity, not optimization, simply because the collective doses involved are likely to be small in most cases, and the costs high.

We have seen that three principles govern decisions for radiation protection, in general. These are justification, optimization, and limitation. Just how useful are they going to be in the case of residual radioactivity? The first, justification, is not particularly helpful for decisions on criteria for residual radioactivity. Presumably, justification decisions were in place long before the need for cleanup of a site for a new use came along. Some take the point of view that no exposure is justified from residual radioactivity because there is no benefit for a practice which has been discontinued. But this is not a useful approach, since obviously it is not practicable to impose a requirement of zero dose. Others (including the ICRP) argue that unless there is a net benefit from cleanup, it should not proceed. That is a judgment not easily achieved or, in most cases, defended. Clearly, in some cases the effort will be great, but necessary in order to achieve acceptable levels of individual risk. And in others careless treatment of natural resources cannot be rewarded by a decision that cleanup is not "justified" because the effort required from the owner is too great compared to the benefit. We have to conclude that justification is not a very useful consideration for establishing cleanup criteria.

Optimization, on the other hand, certainly can be applied. In practice, however, unless the contamination is very long-lived or inexpensive to remove it will not usually lead to individual doses that are acceptably small. Of the three radiation protection objectives, I believe that it will usually be found that limitation is the most important, because we will usually be faced with the need to decide what fraction of the overall limit on dose to an individual it is appropriate to allow residual radioactivity to take up.

Next, we must consider whether there are other applicable or useful concepts or criteria, aside from the general principles for radiation protection. There are several; these are shown in Figure 5. The first is the concept of "acceptable risk" as applied to regulation of chemicals in this country. There is a growing consensus that risks on the order of one in ten thousand to one in a million over a lifetime are appropriate regulatory objectives. Risks larger than these are only acceptable in special situations where extenuating circumstances exist, such as during emergencies.

- Acceptable risk: 10^{-4} to 10^{-6} lifetime
- Groundwater policy:
  - Groundwater classification
  - Drinking water standards
- Below regulatory concern:
  - International guidance
  - NRC Proposals for exempt practices
  - EPA Proposals for low-level wastes

Figure 5. Other Constraints on Cleanup Criteria

The radiation doses that correspond to this range of lifetime risk are disconcertingly low. The range is on the order of 0.04 to four millirems per year, for lifetime exposure. Expressed another way, it is essentially a range of from background to background plus four millirems per year.

A second set of useful criteria may be found in national groundwater policy. These are established in legislation (e.g. the SARA amendments to CERCLA) and contain some very tight and rather specific requirements for cleanup, with respect to groundwater. For example, under current interpretation and practice, if groundwater meets the classification criteria for current or potential use as drinking water (and that means most groundwater) then the applicable standards are those for drinking water. That standard is four millirems per year for man-made radioactivity.

So, with respect to groundwater contaminated as a result of residual radioactivity in the soil, we are back to the same constraints for water as those noted above for the upper end of the range of "acceptable" risk.

A final concept, which may be useful in thinking about criteria for cleanup of residual radioactivity, is that of "below regulatory concern" (BRC). The numbers under consideration for BRC levels of radiation dose are in the same range as those already mentioned above. For example, the IAEA has issued recent guidance on acceptable criteria for exemption. For any specific practice, they recommend individual doses that are on the order of one millirem per year, coupled with a finding that exemption is the optimal choice, i.e., ALARA. They offer the further advice that if the practice involves less than 1 man-Sievert (100 man-rem), then you can assume that it is already the optimal choice, i.e., a further demonstration of ALARA is not required. The NRC is also considering BRC criteria. The levels they are proposing (10 millirem/year) would not, however, satisfy current EPA concepts of acceptable risk, or existing drinking water standards.

We can now summarize the requirements for residual radioactivity cleanup criteria. First, the criteria have to be a fraction of the overall limit for individuals -- that is, a fraction 100 millirems per year. Second, they must optimize protection, considering the totality of current and future health risks in relation to the costs. However, in looking at these health effects, usually

through projection of collective dose from residual radioactivity (even if the collective dose involves the world's population and exposures far into the future), we almost invariably find that this criterion will not be as limiting as the first. (The primary case where collective dose is limiting that I am aware of is Carbon-14, as demonstrated in a study for the IAEA of exemption levels for low level waste by Kennedy in 1987.)

Finally, we will have to satisfy other policy constraints, such as those of acceptable risk and groundwater policy. Based on current practice, these may end up involving the same level of cleanup as is required to satisfy general criteria for BRC, at least at the international level.

I want to make one final point that is often overlooked. We tend to concentrate on dose as the unit of measure by which we express radiation limits, because it is related directly to risk. It is probably a mistake to do this in the case of cleanup criteria for residual radioactivity. Dose limits imply the need for a site-specific assessment of doses to hypothetical receptors for each cleanup. That is expensive and it often will be difficult to achieve without controversy. Further, we do simply not need great precision, at each site, in relating contamination levels to the low levels of risk involved after cleanup. It should be enough to do generic site assessments and then to set standards expressed in units that are directly related to contamination levels.

What we need, then, are simple numerical cleanup criteria that apply to all sites. These must satisfy three different kinds of objectives that may each end up being limiting -- individual risk, population risk, or a national policy objective, as in situations when groundwater may be contaminated. Finally, these numerical criteria should be expressed directly in terms of contamination levels, or total amount of contaminant.

What Should Cleanup Standards Do?

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ABSTRACT

Standards for residual radioactivity on soils and other materials should provide adequate protection for: individuals using the materials or occupying the site, other critical population groups that may be exposed due to their proximity to the site or materials, and the general population at risk from atmospheric transport of contaminants or from ingestion of contaminated water or food. Past analyses have shown that the individual site reclaimer or recycled material use is the critical receptor, but that risks to other components of the public should also be considered. Doses received by site reclaimers or recycled material users are generally proportional to the nuclide concentration (pCi/g) for bulk contamination and nuclide surface concentration (dpm/100 cm^2) for surface contamination. Therefore, standards should limit the concentration of individual nuclides or groups of nuclides. Limiting concentration values should be based on appropriate exposure scenarios and multipathway risk assessments. For example, the ease of nuclide removal or leachability from the material should affect the concentration limits.

INTRODUCTION

This paper addresses the basic requirements for cleanup standards for residual radioactivity. Specifically, the following questions will be addressed in this paper:

1. What segment of the public is the most critical in terms of radiation protection?
2. Should only a dose or risk standard be established?
3. Should standards include limits on concentrations of nuclides?
4. Are multipathway risk assessments necessary for establishing cleanup standards?

Standards for residual radioactivity on soils and other materials should provide adequate protection to the public by ensuring that risks from radiation exposures are at acceptable levels. Separate consideration should be given to individuals using the materials or occupying the site, other critical population groups that may be exposed due to their proximity to the site or materials, and the general population at risk from atmospheric transport of the contaminants or from ingestion of contaminated water or food. Cleanup standards should also contain provisions for reducing potential doses to the public to "as low as reasonably achievable" (ALARA).

In general, for nearly all decommissioning and decontamination problems doses to individuals using the site or using recycled contaminated materials is the critical receptor. However, risks to the other components of the public should also be considered. In particular, these risks are a factor in evaluating whether the cleanup level is ALARA.

DEVELOPING CLEANUP STANDARDS

The major steps in developing cleanup standards, shown in Figure 1, begin by collecting information on contaminated sites: the number of sites, the type and extent of contamination, and the nature of the material that is contaminated. In addition costs to clean up the contamination should be estimated. The increase in costs for cleaning to lower levels of contamination should also be determined.

The next major step leads to the determination of risk or dose performance objectives. These risk/dose limits may be established by consistency with risk/dose limits from related regulations. They also may be developed from a cost/benefit analysis that involves comparing the decrease in risk from the cleanup to the added costs of the cleanup effort for decreasing levels of residual contamination.

After the risk/dose performance standard is determined, then the risk assessments provide the nuclide concentration limits consistent with the performance standard.

It is important to establish performance objectives for cleanup standards. The performance objectives should pertain to a health or risk limitation to the above components of the public and should provide a consistent basis for allowing the effective implementation of the standards. When evaluating doses and risks to populations or individuals in the population, potential exposures in the long term should be considered in addition to potential exposures over the near term.

FACTORS INFLUENCING POTENTIAL DOSES FROM RESIDUAL CONTAMINATION

Many factors affect the extent of risk to individuals from residual contamination levels. The factors, shown in Figure 2, can be grouped into four general categories. The first is the physical and chemical characteristics of the contamination: Is it easily dispersible? Is it easily leached by water? If dispersible, is the airborne component respirable? Questions such as these must be determined before evaluating the dose from a particular contamination level.

The pathway between the waste and humans is another barrier affecting the potential dose. Dust loading in the air, transport, decay and dilution in the groundwater are several pathway-related factors that must be considered.

The environmental barrier is a third factor and is closely related to the pathway component. For example, if the contaminant is ingested via groundwater, does direct consumption of the contaminated water constitute the individual's total intake, or is the contaminated water also used to irrigate crops and water animals that are eventually consumed by humans.

The time of exposure is the next factor that influences doses from a given contaminated site. If the dominant exposure pathway is direct gamma radiation, is the individual in proximity to the contamination for 75 percent of his total time, 25 percent of his total time, or half of his working time?

The final factor that affects doses and risks to humans are the nuclide-specific dose factors, i.e., the dose or dose commitment per pCi inhaled or ingested. Some nuclides with very large dose factors, may constitute an extremely small fraction of the total activity at the site but may dominate the dose to an exposed individual because of the large dose factor.

For the component of the public that involves individuals occupying the site or using the recycled materials, many of the natural barriers identified in Figure 2 are bypassed. In general, the only remaining significant barriers for this situation are the physical and chemical form of the contamination, the exposure time as defined by the exposure scenario, and the dose factors. Because some of the barriers are circumvented, this segment of the public is the critical receptor and generally forms the basis for limits in cleanup standards. Exposures to this segment of the public are also highly dependent of the definition of the exposure scenario. Consequently, doses to inadvertent intruders can vary widely for a given degree of contamination on a site, and it is important to construct meaningful, reasonably conservative exposure scenarios for this component of the public.

For the critical population group or the general population, the doses are directly related to the nuclide inventory, as contrasted with onsite exposures that are directly related to the nuclide concentrations. If the onsite reclaimer is considered in establishing cleanup standards, then the performance objective based on an annual dose or risk can be directly related to individual nuclide concentrations. Since individual nuclide concentrations are also a function of the scenario characteristics, concentration limits serve to standardize the exposure scenario considered for the residual radioactivity. Therefore, specific concentration limits remove a considerable amount of the ambiguity and variability that occurs when standards are based only upon a dose or health risk performance objective. A dose limit is difficult to implement for specific sites. An equivalent nuclide concentration cleanup level is needed by the organization doing the cleanup. If a dose performance objective is the only specific criteria in the standard, ambiguities can arise in defining the appropriate exposure scenario, in defining the characteristics and model for the pathway calculations, and in selecting appropriate food chain and dose conversion factors. An example of the variation that can occur in radionuclide concentration limit that can occur from a specific dose limit is shown in Figure 3. For a dose limit of 25 mrem/yr the limiting Cs-137 concentration can have multiple values depending on the amount of clean cover soil over the contaminated area. As shown in the figure, this concentration limit can vary by four

orders of magnitude. In converting a dose standard to an implementable concentration limit extensive dialogue of necessity takes place between the regulatory body and the site owner. Therefore, limiting the standard to a dose performance objective adds significantly to the site owner's burden, as well as to the regulatory burden. Experience has shown that, in general, site-specific concentration limits are driven to their lowest relevant values if the standard contains only a dose performance objective.

For most radionuclides, multipathway performance assessments of residual contamination reveals that for alpha-emitters the highest doses occur from inhalation of airborne contaminant. For mobile beta-emitters, highest doses occur from water transport and ultimate ingestion. For immobile beta-emitters, highest doses generally occur from biointrusion and onsite agricultural scenarios, and for a few strong gamma-emitters, highest doses occur from direct gamma exposure. Because many of the nuclides can be grouped and their characteristics generalized, concentration limits can be developed for groups of nuclides. In addition, for some industry sectors, the nuclide mix is sufficiently constant that the standard may be implemented by specifying a total activity concentration limit.

SUMMARY

Doses to individuals occupying a contaminated site or using contaminated materials receive the highest doses for a fixed degree of contamination than for offsite individuals or populations. A dose performance standard, therefore, to a critical individual is an appropriate basis for cleanup standards. However, the cleanup standards should go beyond this point and should include nuclide concentration limits for bulk contamination as well as for surface contamination. The nuclide concentration limits should also consider physical and chemical characteristic information. Higher standards should apply to nuclides that are not easily removed and are not mobile unless the gamma pathway dominates the dose. For many circumstances it may be appropriate to apply group nuclide concentrations.

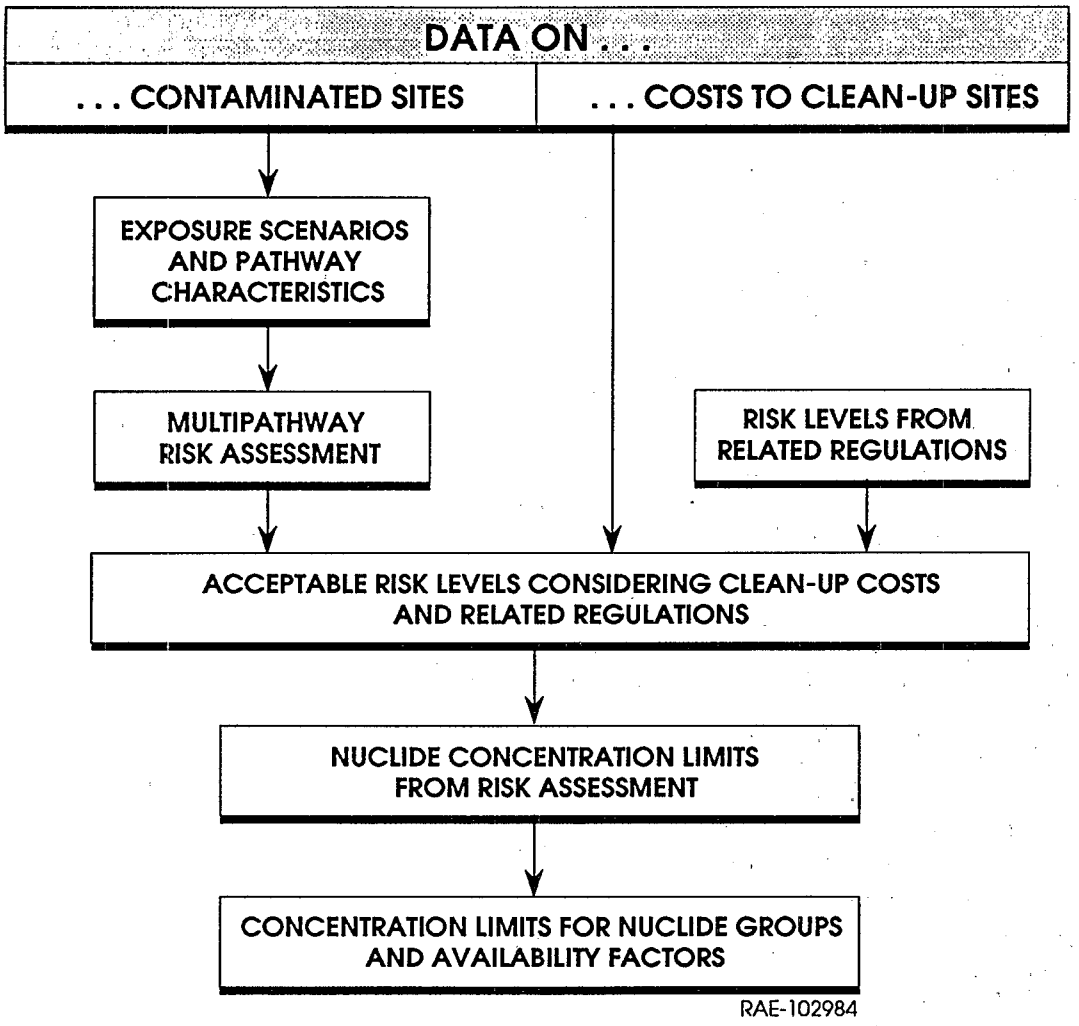

FIGURE 1. STEPS IN DEVELOPING STANDARDS FOR RESIDUAL LEVELS OF RADIOACTIVITY

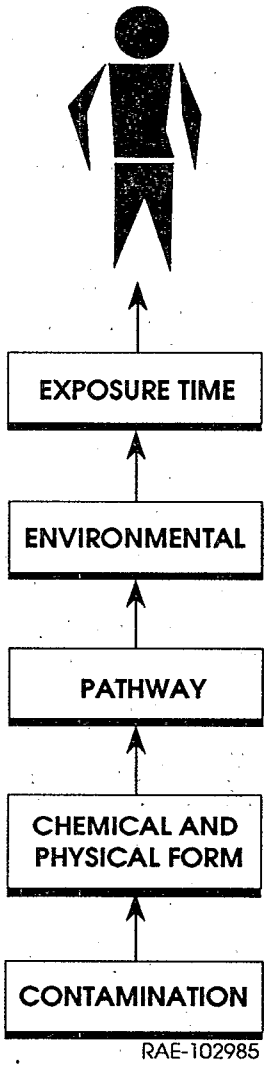

FIGURE 2. BARRIERS BETWEEN MAN AND CONTAMINATION.

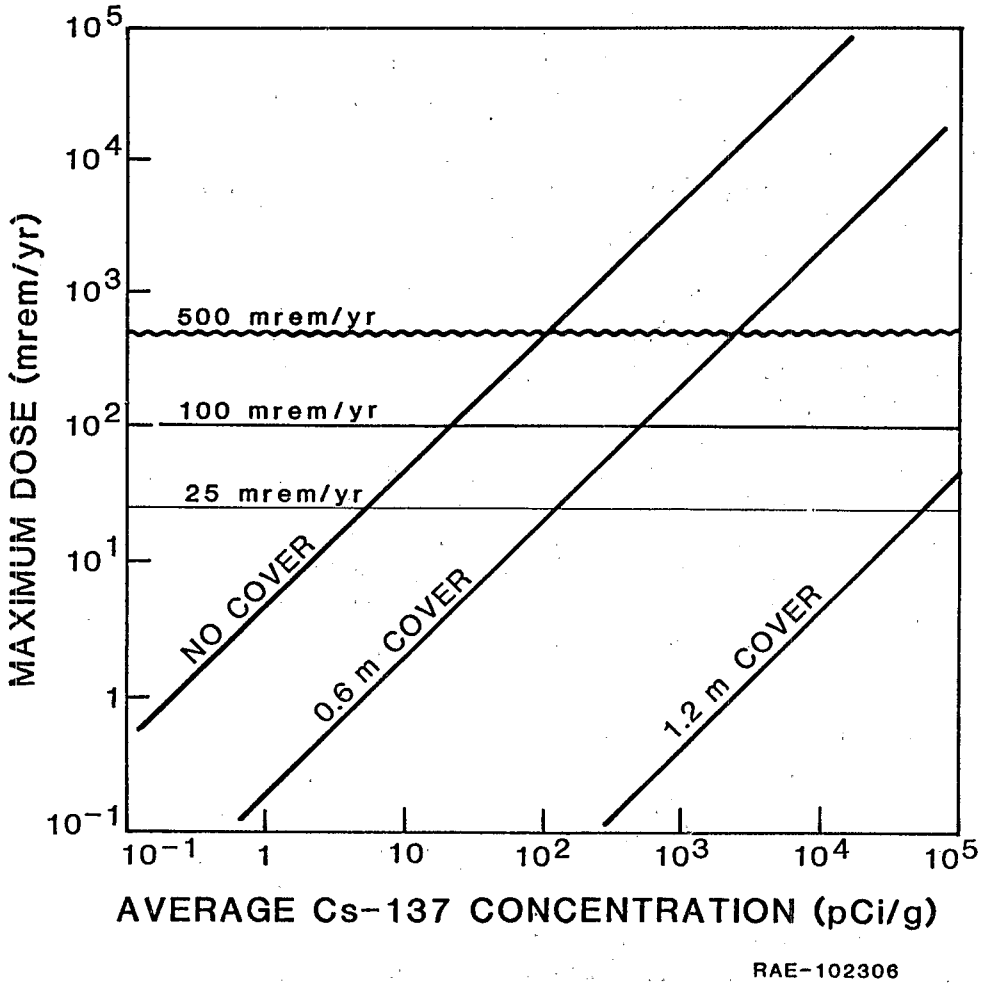

FIGURE 3. RELATIONSHIP BETWEEN DOSE LIMIT AND Cs-137 CONCENTRATION LIMIT FOR SUPERFUND SITE.

Current Status of Residual Radioactivity Criteria in Japan

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ABSTRACT

In this paper, current Japanese regulations concerning residual radioactivity criteria, and efforts to establish such criteria, are reviewed. In addition, the authors comment on desirable forms of residual radioactivity criteria.

Japanese laws and regulations relating to the utilization of atomic energy or radiation do not explicitly mention any residual radioactivity criteria. Although some governmental organizations state policies on reusing nuclear facilities, lands or materials, and efforts to establish criteria have been started, no conclusions have been reached. Considering these circumstances, the authors describe their ideas on desirable forms of residual radioactivity criteria for reusing facilities, lands or materials.

INTRODUCTION

No Japanese regulations concerning the utilization of atomic energy or radioisotopes explicitly refer to criteria for judging the safety or propriety of reuse of nuclear sites, facilities, contaminated materials or radioactive wastes. However, unrestricted reuse of some nuclear facilities has been authorized as a result of judgments based on the current regulations. In this paper, current Japanese regulations implicitly relating to residual radioactivity criteria, and the bases for establishment of the criteria, are reviewed. In addition, the authors comment on the desirable form of residual radioactivity criteria.

REGULATORY SYSTEM OF UTILIZATION OF ATOMIC ENERGY AND RADIATION

The Japanese laws which regulate the utilization of atomic energy and radiation are "The law for the regulations of nuclear material, nuclear fuel material and reactors (Reactor Law)" and "The law concerning prevention of radiation hazards due to radioisotopes, etc. (RI Law)". Facilities that are candidates for reuse comprise two groups, which are regulated separately by these two laws. Facilities in the nuclear fuel cycle, such as reactors or nuclear fuel reprocessing plants, are regulated by the "Reactor Law". Radioisotope handling facilities and accelerators are regulated by the "RI Law".

RULES REGULATING DISMANTLING OF REACTORS AND DISCONTINUATION OF RADIOISOTOPE HANDLING

Reuse of a facility begins after most of the utilization or handling of atomic energy or radiation has ceased. Although a permissible residual radioactivity level at this point is not provided in the above-mentioned laws, some rules for decommissioning are prescribed. The "Reactor Law" provides rules on dismantlement of a reactor, ordering the owner to submit a report to the responsible Minister. The report should cover methods to dispose of nuclear materials and to decontaminate the reactor. If the Minister judges the cleanup methods to be inappropriate, he can order the owner to improve his methods. For example, appropriateness of the residual radioactivity level set by the owner for his dismantled reactor is judged by the Minister. Each case is judged individually (case by case basis).

Similar rules are prescribed for radiation utilization in the "RI Law". Each submitted plan to decontaminate and to set a residual radioactivity level is judged by the responsible Minister.

EXAMINATION CONCERNING ESTABLISHMENT OF RESIDUAL RADIOACTIVITY CRITERIA

In the following sections, the basis for establishing residual radioactivity criteria by governmental organizations is reviewed.

Reuse of decommissioned reactors

The decommissioning of the Japan Power Demonstration Reactor, which was used for research and development work related to power generation, is now in progress. At the start of the decommissioning project in 1985, the Nuclear Safety Commission, one of the advisory organs of the Japanese government, issued a report on its regulatory policy for the decommissioning research reactors.

In the report, the Commission stated two conditions that should be confirmed when the decommissioning process is completed: 1) all of the nuclear materials in the reactor should be removed, and 2) the radioactive wastes generated in the operation should be properly disposed of. The report suggested that the radioactive waste disposal methods should include reuse or recycling of extremely low-level radioactive wastes. However, this report did not refer to any concrete criteria for a residual radioactivity level.

Reuse of low-level radioactive waste repositories

The Nuclear Safety Commission issued a report in 1985, concerning regulatory policy on land disposal of low-level radioactive solid wastes (LLW). In this report, the Commission made suggestions concerning reuse of LLW repositories, and on residual radioactivity criteria for them.

The Commission pointed out that after radioactive materials in the wastes have decayed to a sufficiently low level, restricted reuse may be possible, under certain institutional controls to prohibit or restrict any actions that possibly could pose a hazard to the public.

In addition, the report referred to a stage at which a repository could be exempted from any regulatory control. A repository at that stage could be reused without restoration. According to the report, the primary considerations for exemption of a repository from regulatory control are the levels of radioactivity concentration in the disposed wastes and in the soil of the repository. The report stated that, using an appropriate dose estimation model, residual radioactivity criteria (below which no radiation control should be needed) could be derived from a certain level of the radiation dose to the public.

Reuse or recycling of radioactive wastes

The Nuclear Safety Commission also indicated in the "Regulatory Policy Report" of LLW, a possibility of restricted reuse of extremely low-level radioactive wastes (ELLW). Some examples of ELLW reuse or recycling are the reuse of contaminated concrete chips or blocks as fillers for land development, and of metal piping as construction supplies or raw materials.

The reuse or recycling of radioactive wastes should be regulated on a case-by-case basis, subject to estimation of dose to the reusers from residual radioactivity. Therefore, the primary residual radioactivity criterion for restricted reuse or recycling of ELLW should be a certain dose level to the reusers.

Dose to be exempted from regulatory control

In its report submitted in 1987, the Radiation Council suggested a criterion for unrestricted recycling of radioactive wastes, such as metals, coming from decommissioning the reactors.

With regard to shallow land disposal of LLW, the Council concluded that the criterion for exemption of LLW site should be set so that the estimated dose is sufficiently low compared with the dose limit for the public, i.e., an effective dose equivalent of 1 mSv per year (100 mrem). The criterion should ensure that the public will not be exposed to radiation beyond the dose limit (1 mSv/y), even taking into account the possibility of present and future exposure from other sources and practices.

The Council adopted an individual dose of 10 micro-Sv (1 mrem) per year as the criterion for LLW disposal, in concordance with both the International Commission on Radiological Protection and the International Atomic Energy Agency. The report suggested a similar approach in establishing the criterion for unrestricted reuse or recycling of radioactive wastes.

Investigations for establishment of residual radioactivity criteria

An investigation for establishing residual radioactivity criteria for reuse or recycling of radioactive wastes has been started by the Science and Technology Agency of Japan. Some scientific investigations for this problem are also being carried out by the Japan Atomic Energy Research Institute and some other research organizations.

CONCLUSION AND PRIVATE COMMENTS

As mentioned above, although current Japanese regulations implicitly refer to a residual radioactivity criterion for reuse, no generic criteria have been established yet. No general conclusions have been reached regarding desirable criteria characteristics.

The criteria for reuse of buildings or sites, are considered to be certain dose levels to the reusers. A building or site owner should follow regulatory procedures pertaining to reuse (e.g., the report on decommissioning to the responsible Minister), thereby establishing the operational criteria for residual radioactivity, on a case-by-case basis.

On the other hand, the criteria for reuse or recycling contaminated material or radioactive wastes are the radioactivity concentration levels. Types of reused materials or radioactive wastes, and the ways to reuse them, can be easily identified in streams of reuse. Therefore, it should be practical to set the criteria for each reuse stream, derived from a certain basic criterion, such as a dose limit.

REFERENCES

[1] The law for the regulations of nuclear material, nuclear fuel material and reactors, (1957).

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[3] Nuclear Safety Commission: *Regulatory policy for decommissioning of research reactors*, (1985).

[4] Nuclear Safety Commission: *Regulatory policy for land disposal of low-level radioactive solid wastes*, (1985).

[5] Radiation Council: *Dose to be exempted from regulations concerned with shallow land disposal of radioactive wastes*, (1987).

Establishment of Criteria for the Unconditional Release of the Shippingport Atomic Power Station Site

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ABSTRACT

This paper describes the method used to develop site release criteria for the Shippingport station. It also provides information on initial site characterization, initial radiological conditions, and the application of the DOE document "A Manual for Implementing Residual Radioactivity Guidelines."

Although the direct use of release limits is discussed as an alternate to basing the limits on a cost benefit basis, the paper does show that both the direct use and cost benefit analysis were important in defining the final release criteria.

The development of a procedure which (1) considered the cost of postulated remedial actions and (2) employed an appropriate number of scenarios to demonstrate compliance with applicable limits and the ALARA philosophy is also discussed.

INTRODUCTION

The Shippingport decommissioning project was the first to involve a large-scale, commercial nuclear power plant. Shippingport was also the first nuclear power plant to be decommissioned which had a long period of power operation.

The Shippingport Atomic Power Station was located 25 miles northwest of Pittsburgh, Pennsylvania on the south bank of the Ohio River. The reactor was a four-loop, 72 MWe pressurized water reactor. It was operated by Duquesne Light Company from late 1957 until October 1982 and produced 7.4 billion kwh of electricity.

The reactor and containment were housed in four underground concrete vaults. Two boiler chambers each housed two of the coolant loops. Because of the low radiation levels that

existed in the reactor containment chambers, gross reactor systems decontamination was not required. One of the unique features of the project was the one-piece removal of the reactor pressure vessel. (Figure 1)

The Shippingport project involved the dismantling of reactor equipment from the plant and disassembly of its buildings, as well as removal, transport and disposal of radioactive materials. (Table 1) Irradiated materials were shipped by barge to DOE's disposal site at its Hanford facility in Richland, Washington. (Figure 2) The Shippingport decommissioning project has restored the site to a condition suitable for unrestricted use.

RELEASE CRITERIA: BASIS

DOE's decision to decommission Shippingport was made in August 1982. The site was turned over to GE (the Decommissioning Operations Contractor) in September 1984. At that time the radiological criteria which had to be met before the site could be returned to its owner, Duquesne Light Company, had not been defined. The first step toward definition of the release criteria was made in October 1984 when DOE notified GE that Regulatory Guide 1.86 was to be used to define allowable surface contamination limits. At the same time, GE was asked to perform a sensitivity analysis to estimate the costs to reduce future site doses to the public. We were asked to evaluate 3 levels: 100, 25 and 10
mrem/year.

The second step occurred on Sept. 30, 1985 when DOE issued their working draft of "A Manual for Implementing Residual Radioactivity Guidelines."

INITIAL SITE SURVEYS

An initial surface survey of the site was performed in June 1985. Measurements were made using a scintillation detector held 5 cm off of the ground, and surface soil samples were taken for gamma isotopic analysis. Samples were typically taken where dose rate readings were highest. Only two of the three isotopes (besides tritium) which had been identified in plant systems (Cobalt-60, Cesium-137 and Antimony-125) were found in the soil. Table 2 provides a summary of the concentration of radionuclides found in the soil. Dose rate readings were not taken where radiation from neighboring structures and equipment raised the general background, nor were samples taken in paved areas.

Although the initial site survey data was limited, it was sufficient to perform a preliminary dose assessment the latter part of 1985. A post-decommissioning rough estimate of 22.4 mrem/year to a hypothetical site resident was calculated by summarizing the exposures from the six applicable pathways defined in DOE's draft "A Manual for Implementing Residual Radioactivity Guidelines." This analysis not only provided an initial assessment of the expected dose but also identified "direct exposure" as the predominant exposure pathway. The discovery that 98% of the calculated annual dose would come from direct exposure suggested that cost effective remedial action should be directed at the layers of near surface soil.

The second phase of the initial site survey conducted in late 1985 also determined the degree of on-site concrete contamination. Samples were taken from areas that were known to be contaminated as well as those areas that were thought to be clean.

A hand drill was used to break up concrete over an area of about 100 cm^2 to a depth of not more than 0.6 cm. A total of about 10 grams of concrete was collected for each sample, and a gamma isotopic analysis was made using an intrinsic germanium detector. Table 3 shows the results of the concrete sampling program. As expected, concrete from the Fuel Handling Building canal and from sump areas had the highest concentrations. Cobalt-60 made up more than 99% of the activity detected.

The final part of the initial survey was a subsurface soil sampling program conducted in September 1986. Fifty 10 cm holes were drilled, ranging in depth from 0.5 to 4 meters. Soil was collected in 0.5 meter increments (about 800 grams) and a gamma isotopic analysis performed on site. Table 4 summarizes the concentrations of residual radioactive materials found in the subsurface soil.

SCENARIOS

In early 1986 a program was started to define final site release criteria. At the same time a program was also launched to prepare a cost benefit analysis. In the course of this work, four scenarios were evaluated which covered potential future uses of the site.

(1) Residential: A home is built on the site (including excavation of a basement down to elevation "-3 m"). The family that resides there has a garden, and livestock, and uses water from an on-site well. Particulars about this scenario are defined in detail in DOE's "A Manual for Implementing Residual Radioactivity Guidelines."

(2) Occupancy: A concrete substructure which is open at the top somewhere within the top 3 meters of the site surface has been excavated and turned into an office area. The use is based on an occupancy of 40 hours/week and 50 weeks per year. The minimum size considered to be occupiable is a room having a floor area of at least 10 m^2 and a minimum dimension of 3 m.

(3) Souvenir: A piece of concrete weighing less than 50 kg and located within three meters of the surface is excavated and taken as a souvenir. The exposed person was assumed to live 2 m from the souvenir all year.

(4) Exposed Slab: A monolithic block of concrete not amenable to occupancy, but existing within 3 m of the surface is exposed by excavation. The minimum area considered would be 2 m^2 and minimum dimension 1 m. An individual sits on it for 168 hours while it is exposed.

For completeness, the subsurface concrete not covered by the latter three scenarios is included with the adjacent soil. Thus any contribution to the annual dose from this material is considered via the residential scenario.

COST BENEFIT ANALYSIS

An annual dose was computed for a hypothetical family (residential scenario) establishing a farm on the most contaminated location of the Shippingport site. Figure 3 is a diagram of the farm site showing a dose from residual radioactivity of 3.5 mrem/year.

The cost benefit analysis showed that remedial action that could not be justified on a strict cost benefit ratio could still be undertaken as a matter of prudence because the overall cost was low.

A dose assessment was also made for a worst case "occupancy scenario." It was assumed that the Fuel Handling Building Canal was excavated and converted into an office space. In this case an annual dose as high as 3.82 rem could be received by the office workers.

This analysis showed reduction of the canal dose derived in the worst case scenario (occupancy by office worker) to negligible levels would cost 13.2 man-rem and $898,000. The analysis showed that such high costs should not be expended because the expected benefit would be small.

SITE RELEASE CRITERIA

GE's assessment of the 4 worst case scenarios and cost benefit analysis were then reviewed by DOE. Based on this data, DOE issued its Shippingport site release criteria on January 6, 1987. It was defined as "...100 mrem/year total committed effective dose equivalent to the maximum exposed individual of the general public under the worst case scenario..." In addition GE was directed to apply the philosophy of ALARA to the release of the site.

Adherence to the acceptable surface contamination levels defined in Table 1 of Regulatory Guide 1.86 was established as a goal for the project. Provision was made for DOE concurrence for release at higher levels (but not in excess of the limits), if achievement of the goal should be too costly. Figure 4 gives a time line showing all of the major events leading up to establishment of these criteria.

LIMITING CONDITIONS FOR RELEASE

Radiological conditions which could exist on the site without causing the release criteria to be exceeded were derived based on the four referenced scenarios (Figure 5).

- Top 3 meters of soil: The average concentration must be less than 6 pCi/gram for Co-60. Vertical averaging is limited to 0.15 meters, and horizontal averaging to 100 m^2
- Other soil: The average concentration must be less than 100 pCi/gram Co-60. Horizontal averaging is limited to 100 m^2, vertical averaging is limited to 3 m.
- Occupiable concrete substructure: The average exposure three feet from any wall in the "room" must be less than .05 mrem/hour.

- Souvenir: The average dose rate from a souvenir must not exceed 0.01 mrem/hours at 2 m.

- Exposed slab: The average dose rate at contact (5 cm) must not exceed 0.6 mrem/hour.

Two definitions supplemented these limiting radiological conditions. One placed a restriction on the magnitude and extent of hot spots allowed. The other specified the modifications required when confronted with a mixture of isotopes.

- Hot Spot Criteria: Allowable hot spots can have concentrations up to ten times the average -- provided that the area bounding the hot spot does not exceed the value determined using the following relationship:

$$\text{Maximum Area of Hot Spot} = \frac{100\ m^2\ (\text{Average Concentration})^2}{(\text{Hot Spot Concentration})^2}$$

- Mixtures of isotopes: Limiting concentrations (residential scenario) must be reduced by 4% for each pCi/gram of cesium-137 in the top 3 m of soil and by 2% for each pCi/gram of antimony-125 when cobalt is not the only isotope present. For deeper soil, the reduction would be 1% per pCi/gram of cesium-137 and 0.4 per pCi/gram of antimony-125.

APPLICATION OF ALARA

Reduction of radiation doses to values "As Low As Reasonably Achievable (ALARA)" at Shippingport was required as part of the release criteria. The usual cost ratio of $2000 per man-rem avoided was used to determine whether dose reduction was justified. Actions with cost ratios in excess of this rate were not taken, while actions with lower cost ratios were pursued.

Table 5 summarizes the application of ALARA in the residential scenario. The normal decommissioning effort independent of efforts to apply ALARA would result in a dose of 8.8 mrem/year to a future resident. Installation of a drainage cap already part of the decommissioning program reduces the dose to 3.5 mrem/year.

One other decommissioning activity helped reduce the dose from 3.5 to 2 mrem/year. The top 3 feet of soil in areas formally occupied by buildings was removed.

ALARA in this scenario was met with activities already planned as part of the decommissioning program.

The probability that the Occupancy, Souvenir or Exposed Slab scenarios could become real exposure pathways is low for two reasons. The chance of the required sequence of events occurring is remote, and the real occupancies would be much less than the postulated ones. Therefore, reducing the dose to values less than the limiting ones was not cost beneficial. This is consistent with direction received from DOE which states that action to reduce the expected dose rate from the canal to values less than 100 mrem/year "would result in increased costs and worker exposure... disproportionate to the benefit of further decontamination." GE believes for these cases that the reduction in dose to a value of 100 mrem/year meets ALARA requirements.

APPLICATION OF TABLE 1 TO REGULATORY GUIDE 1.86

Table 1 of Regulatory Guide "Acceptable Surface Contamination Levels" was established as a goal for reducing residual contamination, in this case, on surfaces of concrete left below grade.

In the residential scenario all the subsurface material is considered to be like soil so surfaces have no meaning, and the Table is not applicable.

In contrast, the Table 2 regulatory guide values can be considered to apply to surfaces of underground concrete when it is exposed per the occupancy scenario. There is no direct correlation, however, since the contamination exists not on the surface of the concrete, but mixed in the near surface concrete matrix. This anomaly can be resolved by using a reference measurement. A near surface (0.6 cm) concentration of 100 pCi/gram Co-60 gives a contact reading of approximately 100 ncpm. This corresponds to the reading expected from a smooth surface at 5000 dpm/100 cm^2 (the Table 1 limit). The 100 pCi/g concrete also generates an area dose rate of approximately 0.05 mrem/hour (.01 mrem/hour for each of the walls and the floor). Since this second value is the limiting dose rate for the occupancy scenario, achievement of the 100 mrem/year value also demonstrates equivalent achievement of the Table 1 goal.

Limitation on averaging in Table 1 to areas of 100 cm^2 and 1 m^2 was not applied since the exposure received according to the occupancy scenario is from the general habitation of the room. Compliance was demonstrated instead by requiring the average dose rate in the occupiable portions of the room to be less than 0.05 mrem/hour. This reduced the effort required to prove that the criteria was met without reducing the assurance that the 100 mrem/year dose rate limit would not be exceeded.

The souvenir scenario and the exposed slab scenario were treated differently. Each represents a case where the probability is small that the actual use will correspond to the identified one. In addition, it is unlikely that the occupancy assumed in the two scenarios would be as extensive as that modeled. Therefore, the expected dose rate will be much less than the limiting one (100 mrem/year) and it is "unlikely to result in an unreasonable risk to the health and safety of the public."

Therefore cleaning miscellaneous pieces of rubble and nonoccupiable trenches beyond the specified limits (100 mrem/year by the applicable scenario) would not be cost beneficial.

SUMMARY

Site release criteria was generated for the Shippingport Station Decommissioning Project.

An implementation plan translated the criteria into limiting conditions for the site. This plan demonstrated that ALARA criteria were met, and that the Table 1 limits in Regulatory Guide 1.86 were reached.

The annual dose to a member of the public via the most restrictive probable scenario (the residential scenario) after decommissioning would be less than 5 mrem. In addition, extra costs above and beyond the applied decommissioning procedures to achieve this low dose were less than $30,000. The Shippingport experience should be considered by other nuclear facilities as an incentive to keep operations under strict control. When unnecessary spread of contamination can be avoided, the costs of decommissioning (both financial and radiation exposure) can be minimized.

TABLE 1

SHIPPINGPORT STATION DECOMMISSIONING PROJECT

KEY STATISTICS

- Reactor Vessel Package - 900 Tons
- Radwaste Volume - 3000 Cu Yds
- Radioactive Contents - 14,500 Curies
- Vessels/Tanks - 130
- Chamber Steel - 22,400 Tons
- Contaminated Concrete - 50 Cu. Yds
- Non-Contaminated Rubble - 15,000 Cu. Yds
- Contaminated Pipe - 56,000 LF
- Non-Contaminated Pipe - 55,000 LF
- Asbestos Waste - 500 Cu. Yds

TABLE 2

SUMMARY OF SURFACE SURVEY RESULTS

Isotope	Frequency Of Detection (% of Samples)	Distribution of Detectable Activity (pCi/g) *		
		Maximum	Median	Mean
Co-60	64.3	68** 5.3	0.69	1.04
Cs-137	85.4	3.5	0.39	0.52
Sb-125	0	---	---	---

* Only listed for those samples having statistically detectable activity

** One sample (68 pCi/g) contained a resin bead so the second highest concentration is also included. The initial value was not included in the average.

TABLE 3

SUMMARY OF CONCRETE SURFACE SURVEY RESULTS

Area	# of Samples	Surface Activity (pCi/g) Range	Median
B/D Boiler Chamber	15	1.3 to 5.9	2.2
A/C Boiler Chamber	15	8.0 to 591	4.1
Fuel Handling Building Canal	39	88 to 35,500	6,820

TABLE 4

SUMMARY OF SUB-SURFACE SURVEY RESULTS

Isotope	Frequency Of Detection (%of Samples)		Distribution of Detectable Activity (pCi/g)*		
			Maximum	Median	Mean
	All Samples**	All Holes			
Co-60		47	94	4.00	0.220.32
Cs-137		44	100	1.58	0.210.25
Sb-125		1	4	3.06	1.961.85

* Only listed for those samples having statistically detectable activity.

** Usually there were 16 samples per hole.

TABLE 5

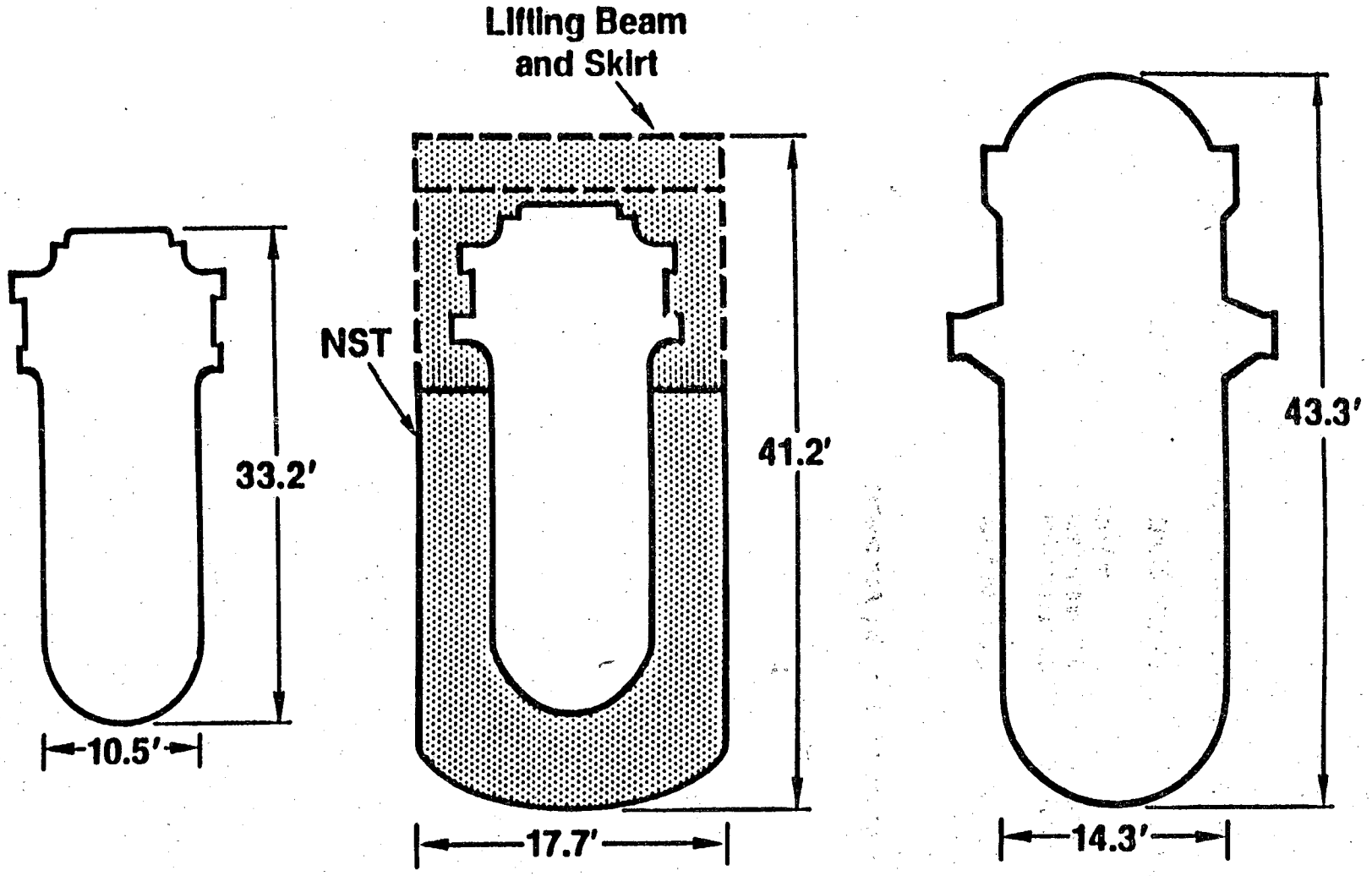
APPLICATION OF ALARA USING THE RESIDENTIAL SCENARIO

Category	Annual Dose*	ALARA
Limit	100 mrem/year	N/A
Site (As Is)	8.8 mrem/year	N/A
Site (Per Plan)	3.5 mrem/year	Yes
Site With Soil Removal	2.0 mrem/year	LTRA **

* 1990 mrem

** LTRA - Lower Than Reasonably Achievable

FIGURE 1



PWR
Shippingport
72 MWe

PWR
Shippingport
with
Neutron Shield Tank,
Lifting Beam and Skirt

PWR
San Onofre 2/3
1100 MWe

FIGURE 2

SHIPPINGPORT STATION DECOMMISSIONING PROJECT

RPVI/NST PACKAGE AND OTHER COMPONENTS ON BARGE

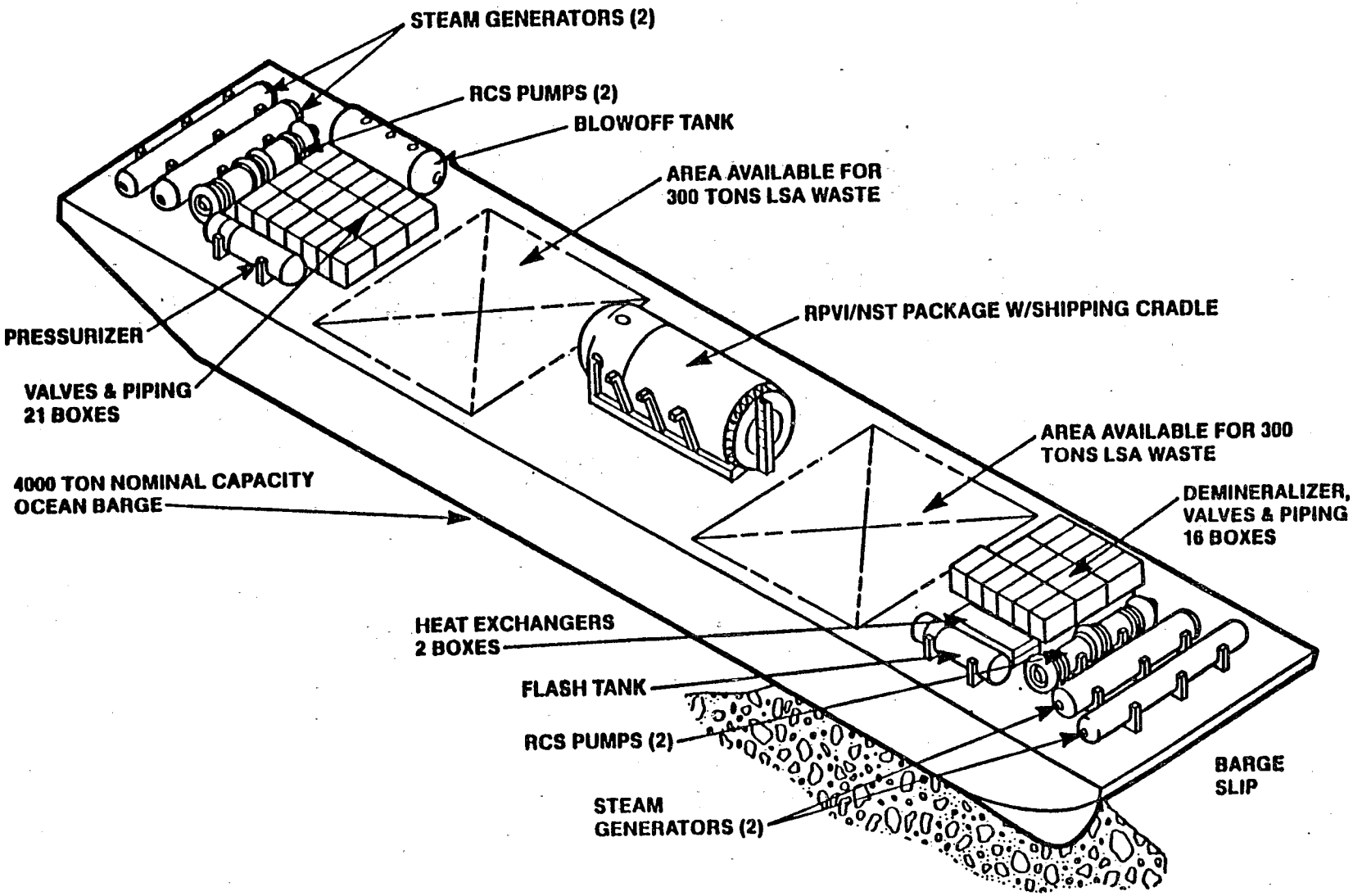


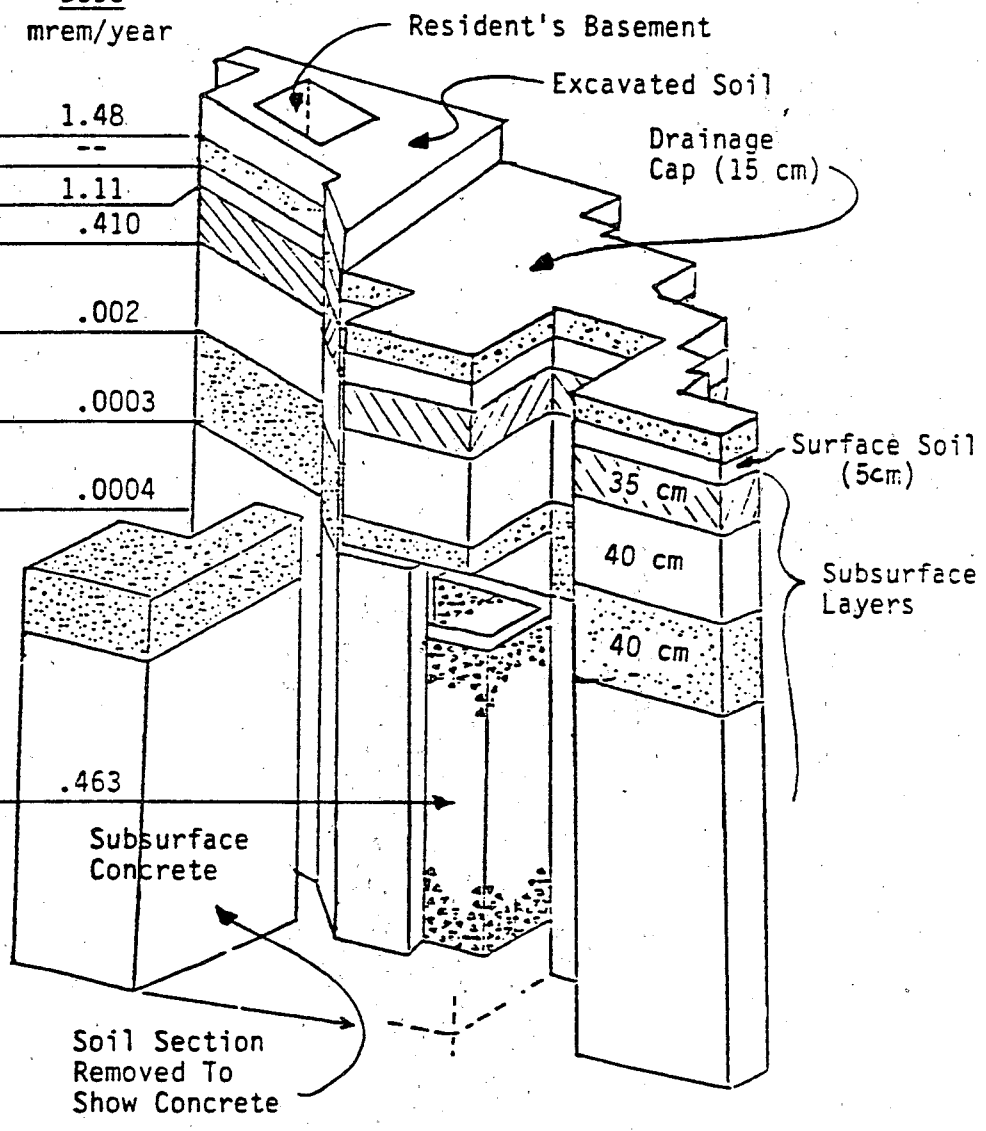
FIGURE 3

FARM-SITE SHOWING DECOMMISSIONINED

BUT UNREMEDIATED CONFIGURATION AND THE

RESULTANT DOSE VIA THE RESIDENTIAL SCENARIO

Soil Concentration (pCi/g)		Dose
Co-60	Cs-137	mrem/year
0.11	.085	1.48
--	--	--
1.0	.96	1.11
.29	.16	.410
.094	.081	.002
.060	.047	.0003
.063	.055	.0004
100* or 100*		.463



*Limiting Concentration Assumed

Total Dose 3.5 mrem/year

FIGURE 4

ACTIVITIES LEADING TO THE

IMPLEMENTATION OF SITE RELEASE CRITERIA FOR THE

SHIPPINGPORT ATOMIC POWER STATION

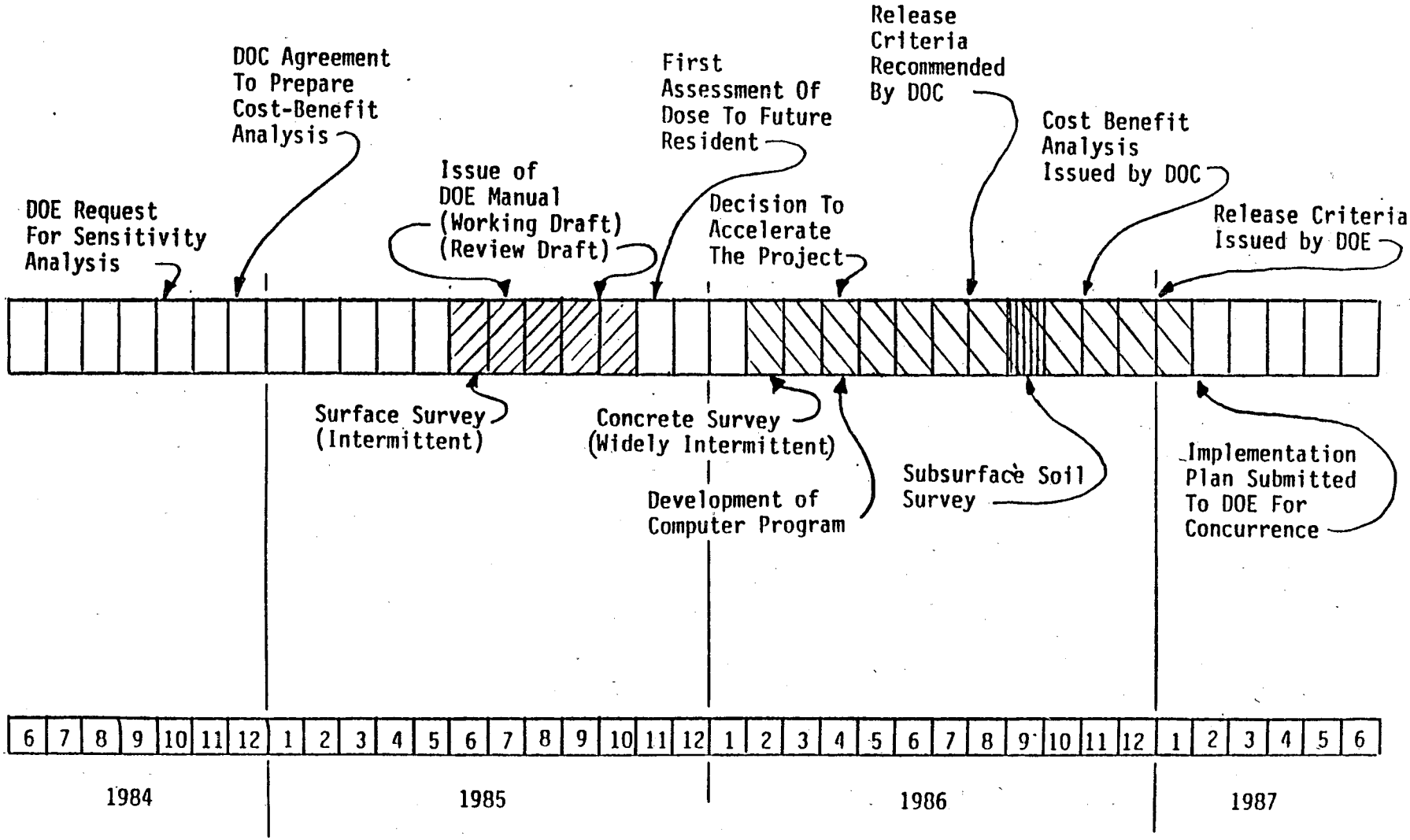
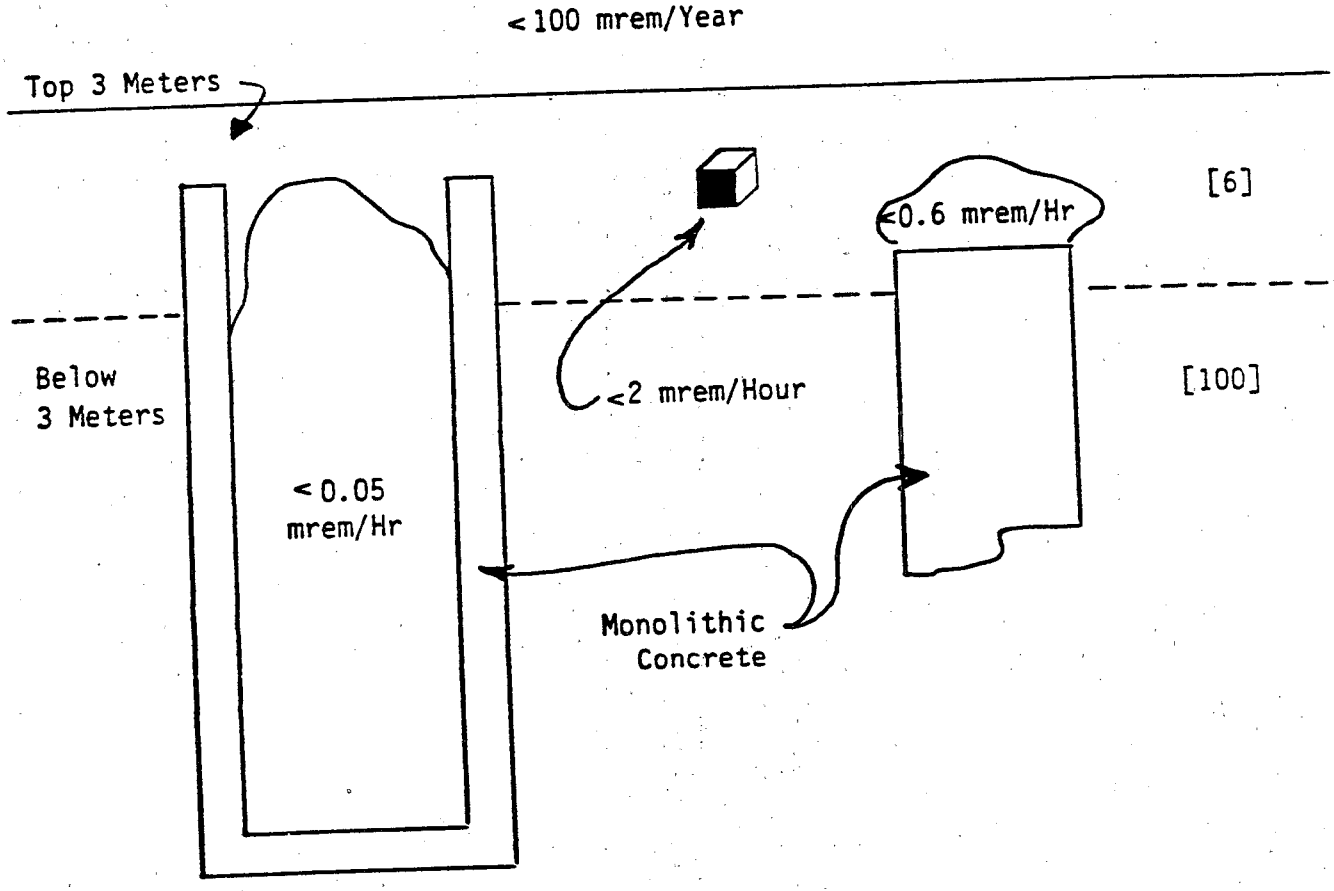


FIGURE 5

LIMITING CONDITIONS
PROPOSED FOR THE RELEASE OF THE
SHIPPINGPORT STATION DECOMMISSIONING PROJECT SITE



^{60}Co Concentration (pCi/g) = [ ]

NRC Residual Contamination Criteria

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ABSTRACT

The Nuclear Regulatory Commission (NRC) is currently addressing the issue of residual contamination criteria. A Commission Policy Statement on exemptions from regulatory control is expected in the very near future. Based on this policy NRC staff will develop interim guidance on acceptable levels of residual contamination early in 1990.

CURRENT CRITERIA FOR REACTORS

The current guidance for terminating nuclear reactor licensees is found in Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors" [1]. This regulatory guide provides methods and procedures considered acceptable by the NRC staff for reactor license termination. It also contains decontamination guidance for release for unrestricted use. This regulatory guide applies to research, test, and power reactor license terminations.

Prior to terminating a license and releasing the site for unrestricted use Regulatory Guide 1.86 recommends that the licensee should --

a. Make a reasonable effort to eliminate residual contamination.

b. Not apply coverings to radioactive surfaces of equipment and structures until the contaminated levels are below those in Table I of Regulatory Guide 1.86.

c. Determine the radioactivity in the interior surfaces of pipes, drain lines, or ductwork by making measurements at all traps and other appropriate access points if the contamination can be shown to be representative of the actual contamination. Inaccessible points on structures, equipment, or scrap should be considered contaminated to levels in excess of the limits for unrestricted release.

d. Make a comprehensive radiation survey.

The NRC staff, and previously the Atomic Energy Commission staff, have used the surface contamination limits given in Table I of Regulatory Guide 1.86 for over twenty years. Table I of Regulatory Guide 1.86 is presented below. These limits were developed based on the variations in natural background and the lower limits of detection of survey instruments. These criteria are not based on a dose objective that relates nuclide concentration levels to exposures to the public. This situation has led to the need for criteria based primarily on individual dose that is directly related to public health and safety.

In addition to the structure and equipment surface contamination levels in Regulatory Guide 1.86, the NRC staff has applied a limit for gamma-emitting nuclides of 5 uR/hr above background as measured at 1 meter from the surface applicable to radioactive material other than surface contamination [3]. Assuming a reasonably conservative occupancy time of 2,000 hours/year, the maximum exposure to an individual would be 10 mRem/yr.

TABLE I
ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDE[a]	AVERAGE[bc]	MAXIMUM[bd]	REMOVABLE[be]
U-nat, U-235, U-238 and associated decay products	5000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1000 dpm α/100 cm^2
Transuranics, Ra-226 Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/ 100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others as noted	5000 dpm $\beta\gamma$/100 cm^2	15,000 dpm $\beta\gamma$/100 cm^2	1000 dpm $\beta\gamma$/100 cm^2

[a] Where surface contamination by both alpha- and beta-gamma emitting nuclides exists, the limits established for alpha- and beta-gamma emitting nuclides should apply independently.

[b] As used in this table dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency and geometric factors associated with the instrumentation.

[c] Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d] The maximum contamination level applies to an area of not more than 100 cm^2.

[e] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

Regulatory Guide 1.86 is currently undergoing revision to make it compatible with the license termination procedures in the new decommissioning rule published in the Federal Register on June 27, 1988 [2]. The revised regulatory guide will reference updated residual contamination criteria. It is expected to be issued for comment after the interim residual contamination limits are issued early in 1990.

NEW CRITERIA DEVELOPMENT

The NRC is currently a member of an interagency task force to develop standards for residual criteria. However, this group is not expected to complete its work until the mid-1990's. In the interim the NRC staff recognizes the need for guidance in this area. The Commission has committed to providing this guidance by December 1989. This guidance is directly related to NRC activities in the areas of Below Regulatory Concern (BRC) waste disposal practices and other regulatory decisions involving the exemptions of radioactive material from regulatory control. The following discussion presents important background information that will lead to the development of residual contamination criteria.

Section 10 of the Low-Level Radioactive Waste Policy Amendments Act of 1985 [4] required that, within six months, the NRC establish standards and procedures, and the technical capability to act in an expedited manner on petitions to exempt specific waste streams from regulation. NRC responded with three actions.

First, On August 29, 1986 the NRC published in the Federal Register [5] a Commission Policy Statement and Staff Implementation Plan. These two documents provide guidance to potential rulemaking petitioners outlining the decision criteria the Commission intends to use to expeditiously process BRC waste stream petitions.

Second, the IMPACTS-BRC computer code for calculating radiological impacts from unregulated disposal was adapted for personal computer use and a draft user guide was published in July 1986 (Volume 2 of NUREG/CR-3585) [6]. Subsequently, the NRC staff contracted with Sandia National Laboratory for technical assistance to critique, validate, and verify the computer code.

Third, on December 2, 1986 the NRC published in the Federal Register an advanced notice of proposed rulemaking (ANPR) [7] requesting comments on the development of a generic BRC level for wastes. Over 90 comments were received in response to the ANPR reflecting diverse views on how the NRC should proceed. Many commenters opposed the concept of any level of radioactivity being BRC and others urged NRC to proceed promptly on the generic rulemaking. In March 1988 the Commission delayed the rulemaking and directed the staff to first prepare for Commission consideration options for a broad policy statement that establishes a generic limit for exposures that are below regulatory concern.

The policy statement addresses exemption decisions as a whole, not only those BRC issues for waste management, but also licensing applications for consumer products, existing exempt quantity limits, decommissioning, and effluent releases. This policy statement would provide for more efficient and consistent regulatory actions in connection with exemptions from specific NRC requirements. A draft policy statement was prepared and discussed at the International Workshop on Rules for Exemption from Regulatory Control sponsored by the NRC and the Nuclear Energy Agency in October 1988. An advance notice of a policy statement was issued for public comment in the Federal Register on December 12, 1988 [8]. The Advance Notice recommended a 10 mRem/yr individual dose criterion as one basis for establishing a floor for curtailment of ALARA activities. This value considered optimum use of Commission resources to address matters of radiological protection, the variations in background, risk perceptions, BRC versus de minimis distinctions, the linear non-threshold hypothesis, and practical implementation.

The policy and the comments received are currently being considered by the Commission. We expect the Commission to take action in the very near future.

Based on the Commission action and dose objectives that are set, the NRC staff will prepare interim criteria on residual contamination early in 1990. Included will be guidance on residual contamination levels on a nuclide-by-nuclide basis. These nuclide-by-nuclide data will be developed from pathway analyses performed under a contract with Pacific National Laboratories. These analyses are based on direct exposure, ingestion, inhalation, and groundwater pathways.

REFERENCES

[1] U.S. Nuclear Regulatory Commission Regulatory Guide 1.86, *Termination of Operating Licenses for Nuclear Reactors,* June 1974.

[2] *General Requirements for Decommissioning Nuclear Facilities,* Final Rule, Federal Register, Vol. 53, No. 123, pp. 24018 - 24056, June 27, 1988.

[3] *Letter to Dr. Roland A. Finston,* Stanford University, from John A. Stolz, NRC, March 17, 1981.

[4] *Low-Level Radioactive Waste Policy Amendments Act of 1985,* Public Law 99-240, January 15, 1986.

[5] *Radioactive Waste Below Regulatory Concern; Policy Statement,* Federal Register, Vol. 51, No. 168, pp. 30839 -30847, August 29, 1986.

[6] Forstom, d.M., Goode, D.J, *De Minimis Waste Impacts Analysis Methodology,* NUREG/CR-3595, Volume 2,-July 1986.

[7] *Radioactive Waste Below Regulatory Concern; Generic Rulemaking,* Federal Register, Vol. 51, No. 231, pp. 43367 - 43369, December 2, 1986.

[8] *Policy Statement of Exemptions from Regulatory Control,* Federal Register, Vol. 53, No. 238, pp. 49886 - 49891, December 12, 1988.

Status and Implementation of the NRC Policy on Exemptions from Regulatory Control

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ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is currently considering a broad policy on exemptions from regulatory control. An advance notice of policy development was published for public comment in December 1988, and an international workshop and public meeting were held in October 1988 and January 1989 respectively. The policy will establish the framework within which the NRC will consider specific regulations and licensing actions that exempt certain practices from all or part of the normal system of regulatory control. Included within the policy will be the specification of numerical individual dose criteria that define levels below which the Commission believes that further compliance with the ALARA principle is unwarranted. The policy will have broad applicability in areas such as waste disposal, decommissioning, and consumer products.

The NRC also is developing, as part of the initial implementation of the policy, interim guidance on residual contamination criteria for soils and structures that correspond to the individual dose criterion. This guidance will probably take two different forms. First, values of dose per unit concentration for exposure pathways such as inhalation, secondary ingestion, and direct radiation will be provided for various radionuclides to facilitate site-specific analyses. Second, a generic release for unrestricted use scenario will be evaluated and the concentration of each radionuclide corresponding to the individual dose criterion tabulated.

Beginning in early 1988, the U.S. Nuclear Regulatory Commission (NRC) began development of a broad policy on Exemptions From Regulatory Control. This effort was in addition to the actions already underway to implement the Low-Level Radioactive Waste Policy Amendment Act of 1985 which directed the NRC to develop procedures for determining the types and quantities of radioactive wastes that could be considered to be below regulatory concern. The thrust of the broad policy, however, is to establish a framework within which the NRC can

consider all types of exemption, including consumer products, decontamination and decommissioning for unrestricted use, and potentially the recycle of radioactive materials.

The purpose of a broad policy statement is to establish the basis upon which the Commission may initiate the development of appropriate regulations or make licensing decisions to exempt certain practices from some or all regulatory controls. While we envision this policy to be directed principally toward rulemaking activities, it may also be applied to license amendments or license applications involving the release of licensed radioactive material either to the environment or to persons who would be exempt from the NRC's regulations.

In December 1988, an Advance Notice of Proposed Policy Development was published in the Federal Register which contained a proposed policy statement and solicited comment on the approach being considered for exemptions. The NRC received over 200 comment letters from members of the public, public interest papers, the industry, and various state, federal, and local governmental organizations. The NRC also solicited comments during a public meeting in January 1989 and the international perspective on exemptions during an international workshop on exemptions held in October 1988. As a result of these comments and the information gathered during the meetings, a revised policy statement was prepared by the NRC staff and submitted for Commission consideration in June 1989. The Commission is presently considering the staff proposal.

The proposed policy would establish numerical criteria for individual and collective dose that would define a region in which exemption of a practice from regulatory controls should be a rather straightforward undertaking. However, a practice resulting in individual or collective doses in excess of the numerical criteria would not automatically be excluded from consideration for exemption. Instead, further analysis would be required to determine if an exemption was the appropriate regulatory approach in that particular situation.

The structure of the proposed policy statement was very similar to the recommendations of the International Atomic Energy Agency (IAEA) in their Safety Series No. 89. However, although fundamental principles of radiation protection were considered in recommending the individual and collection dose criteria, the specific values selected represented a policy judgment based on risk and resource allocation considerations. As a result, the criteria do not exactly agree with the bases for, or magnitudes of, similar criteria selected or under consideration nationally by the U.S. Environmental Protection Agency (EPA), by other countries, or by international agencies such as the IAEA.

It is important to note that, in this policy, the NRC does not assert an absence or threshold of risk at low radiation dose levels but rather establishes a baseline where further government regulation to reduce risks is unwarranted. The presence of natural background radiation and variations in the level of this background are used to provide a perspective on which to judge the relative significance of the radiological risks involved in the exemption decisionmaking process.

A major consideration in exempting any justified practice from some or all regulatory controls hinges on the general question of whether or not application or continuation of regulatory controls is necessary and cost-effective in reducing dose. To determine if exemption is appropriate, the Commission must determine if one of the following conditions is met:

1. The application or continuation of regulatory controls on the practices does not result in any significant reduction in dose received by individuals within critical groups and by the exposed population; or

2. The costs of the regulatory controls that could be imposed for dose reduction are not balanced by the commensurate reduction in risk that could be realized.

The basic numerical criteria contained in the proposed policy statement are illustrated by Figure 1. The individual dose criterion was established at 10 mrem (0.1 mSv) and a collective dose criterion set at 500 person-rem (5 person-Sv). Each of these criteria applies to a practice which is defined in such a way so as to avoid fractionation or deliberate dilution of materials that should otherwise be controlled. If the individual and collective doses from a practice were to fall within the criteria, then, under the proposed policy the practice would be considered as meeting the basic conditions for exemption.

One facet of the proposed policy which has not been finalized is the way in which the potential for exposures to multiple exempted practices is to be handled. The Commission is considering several mechanisms to deal with this issue, including a case-by-case evaluation of other existing exemptions to assure that exposure to several exempted practices will not result in doses which are a significant fraction of the dose limits for members of the general public. A second possibility is to reduce the individual dose criterion if the practice under consideration has the potential for being widespread within the population, such as for consumer products.

The exemption policy does not constitute regulatory permission to transfer radioactive materials to an uncontrolled status. The policy only describes the criteria and framework under which the NRC will take specific rulemaking or licensing actions. The NRC staff anticipates that the typical approach will be for an outside group, such as the nuclear industry, to petition the NRC for an exemption action. In fact, the Nuclear Utility Management and Advisory Council (NUMARC) is planning to submit such a petition in the near future to consider certain waste streams from commercial nuclear power plants as below regulatory concern. The staff will consider such petitions, and, if warranted by the technical evidence, publish a proposed rule for public comment. NRC action on any final rule would then be based upon the proposed rule and the public comments received.

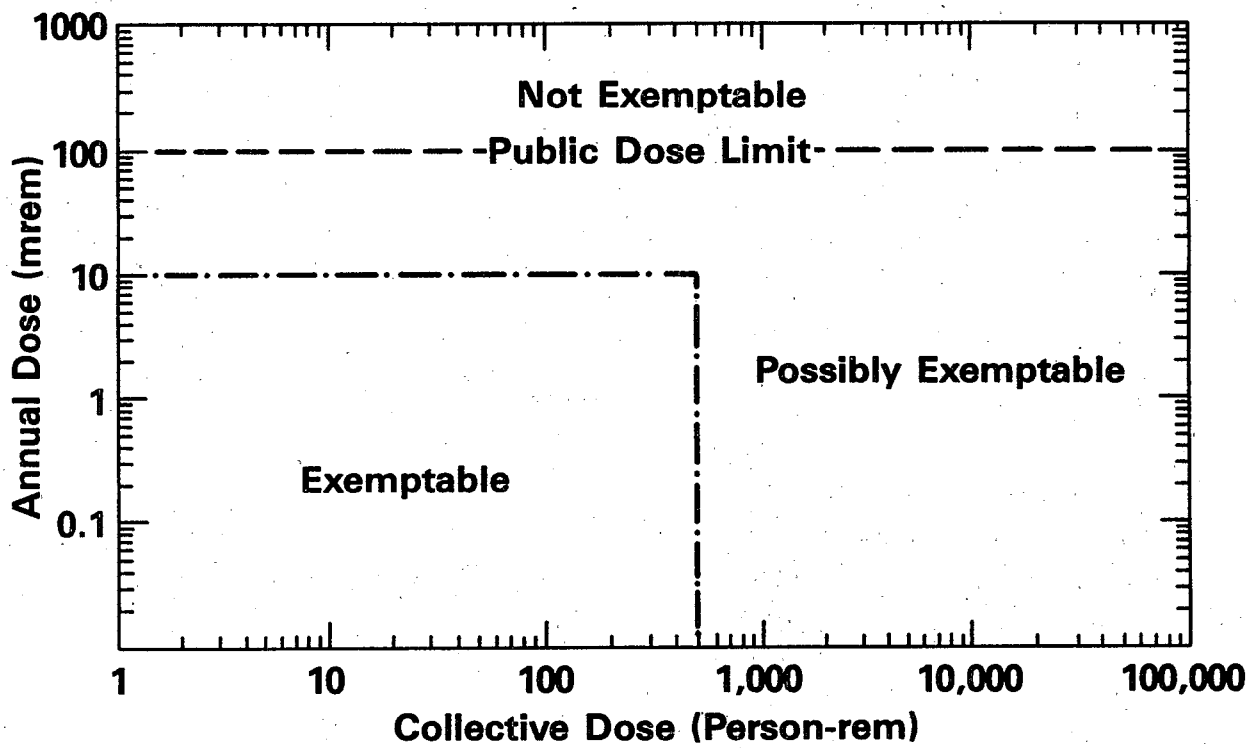
One area in which the NRC recognizes the need for additional information to translate the proposed exemption policy into actual regulatory decisions is the area of residual contamination criteria. The NRC staff, with contractual support from PNL, is currently developing interim criteria for soils and structures. The NRC staff envisions that the criteria will be presented in two formats to facilitate use by NRC, licensees, and other groups. First, the various pathways that could result in exposure, including direct radiation, inhalation, secondary ingestion, food pathways, and groundwater will be examined and the dose per unit concentration for soils or structures tabulated for a variety of radionuclides. We believe that this type of information is essential to a site-specific analysis of potential doses resulting from a decontamination or decommissioning action.

The NRC staff also envisions the need for generic values for each radionuclide that would correspond to a dose equal to the individual dose criterion of the exemption policy. These generic values could be replacements for the guidance currently available in Regulatory Guide

1.86 and the NMSS Branch Technical Position on Disposal Onsite Storage of Uranium and Thorium in Soils. Therefore, the staff expects to develop a combination of exposure pathways into a generic "release for unrestricted use" scenario and to develop values for each radionuclide. A licensee could then simply demonstrate that the residual radioactivity levels on a site are less than the corresponding guidance values and thus show that the dose would be less than the exemption dose criterion. The NRC actions could then focus upon verification of residual radioactivity levels, rather than upon the appropriateness of the pathways analysis.

In summary, the NRC is currently developing both a broad policy on exemptions from regulatory control and interim guidance on the appropriate levels of residual radioactivity for decontamination. Although the NRC is not specifically considering the recycle of materials at this time, decisions on recycle most likely would also be covered within the criteria of the exemption policy.

PROPOSED EXEMPTION POLICY FOR A JUSTIFIED PRACTICE



Surface Contamination Criteria for Free Release

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ABSTRACT

Recycling of materials, equipment, and facilities during decontamination, decommissioning, or remedial action projects requires criteria for surface radioactivity guides. A consensus standard developed by the Health Physics Society on permissible limits of residual surface radioactivity for unrestricted release has gone through an evolutionary process during the last 18 years.

Brief history of surface contamination limits from 1974 through 1989 is presented. Comments on the practical application, limits, measurement methods, and hazardous analysis relating to the limits are reviewed.

The process to develop a consensus standard in the United States on permissible limits of residual surface radioactivity on materials, equipment, and facilities to be released for uncontrolled use began in September 1971 with the formation of a subcommittee of the Health Physics Society Standards Committee (HPSSC). The standard has thus gone through an evolutionary process of over 18 years, including the usual subcommittee deliberations, meetings of the working group with HPSSC, with regulatory agencies, and with industry representatives, reviews of the literature, and consideration and responses to comments and criticisms in several ballots (Shapiro, 1980). In 1974 the subcommittee completed a proposed draft as ANSI N328 (1974), and it was formally sent to American Nation Standards Institute (ANSI) Committee N13. ANSI published the standard as a draft for comment ANSI N13.12. However, N328 was adopted by the NRC in the form of Regulatory Guide 1.86. Subsequently the Health Physics Society Standards Committee working group drafted revisions of ANSI N13.12. Proposed ANSI N13.12 has never been published as an ASNI Standard. In 1983 DOE published ORO31 as an adaptation of the proposed N328. DOE later dropped the N328 version in favor of the Regulatory Guide 1.86 version and issued a final adoption of 1.86 in July 1985.

PROPOSED ANSI N328 (1974)

The limits selected by the N328 working group were indirectly tied to MPC values but it was made clear that these limits were by a consensus based on what appeared to be safe and practical with the existing technology. The limit for Sr-90 was arbitrarily taken as 1000 dis/min per 100 cm^2 because it was approximately the value of background Sr-90 contamination produced by fallout from past above ground nuclear weapon tests. In addition, the Committee decided to use 5000 dis/min per 100 cm^2 as an upper limit for group 2 radionuclides, since higher values could lead to unnecessary high direct radiation exposure. Nuclides with maximum beta energies less than 150 keV were not considered. An abbreviation of the table of limits from N328 is shown in Table 1.

REGULATORY GUIDE 1.86 (1974) AND NRC (1982)

An abbreviated version of the table of limit values from Regulatory Guide 1.86 (1974) and NRC (1982) is shown in Table 2. The only significant difference between the 1.86 (1974) table and the NRC (1982) table is that the NRC (1982) table has a footnote which states: "The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h at 1 cm and 1.0 mrad/h at 1 cm respectively, measured through 7 milligrams per square centimeter of total absorber."

DOE GUIDELINE (JULY 1985)

The DOE (1985) table of limits is identical to the Regulatory Guide 1.86 (1974) table. It includes the NRC (1982) footnote on average and maximum dose rates of 0.2 and 1.0 mrad/h at 1 cm. The ORO 831 (1983) guidelines were essentially identical to N328 (1974) except that a footnote was added which stated: "In the event of the occurrence of mixtures of radionuclides, the fraction contributed by each constituent to its own limit shall be determined, and the sum of the fraction shall be less than 1."

DRAFT ANSI N13.12

An abbreviation of the table of limits in ANSI N13.12 (1978) is shown in Table 3. ANSI N13.12 (1978) is different from N328 (1974) in that "nondetectable" was used for Group 1, and "nondetectable By" and 2000 were used for Group 2. An abbreviation of the table of limits in proposed ANSI N13.12 (1981) is shown in Table 4. The limits in proposed ANSI N13.12 (1981) essentially returned to those of N28 (1974) except that the total value for Group 2 was changed from 1000 dpm/100 cm^2 to 5000 dpm/100 cm^2 and the iodines were moved to Group 2. The 1983 revision of proposed N13.12 was identical to proposed N13.12 (1981) except that it excluded depleted uranium, U-228 and Th-232 from Group 1. Table 5 is an abbreviation of the table of limits in proposed ANSI N13.12 (1983).

The most recent version of the proposed standard ANSI N13.12 (1988) is shown in Table 6. The most significant modification from the 1983 version is addition of a Group 4 that includes natural, depleted, and enriched uranium to less than 10% U-235, Th-232, and their decay products. The removable limit is 200 dpm/100 cm^2 of gross alpha disintegration and the fixed plus removable limit is 1000 dpm/100 cm^2 of gross alpha disintegration.

PRACTICAL APPLICATION

The standard presents requirements and suggestions for measuring and sampling to obtain a complete assessment of the surface being examined. Both the direct and indirect (wipe) monitoring methods are required to be used. Acceptable practices for the purposes of the standard are described for direct alpha and beta activity monitoring and indirect surface monitoring.

The direct alpha activity monitoring requires the distance from the detector window to the surface being monitored to not exceed 0.5 cm, the total absorber of the detector window and the interviewing medium shall not exceed 2 mg/cm^2 and the scanning speed must be slow enough to ensure a detection frequency of 50% or more at the guide level. Table 7, taken from Table B-1 of ANSI 1988, lists the frequency or probability of observing "n" or more counts from a point source of 300 d/minute for several efficiencies and counting times. For direct monitoring of betas the detector window and intervening medium must transmit at least 50% of the incident beta particles, the detector window must be no more than 1.5 cm from the surface, and the scanning speed must be slow enough to ensure a source detection probability of 50% or more.

For indirect (wipe) monitoring method the instruments must have sufficient counting efficiency so that the average limiting alpha or beta disintegration rates could be determined with an accuracy of better than ± 50% at the 90% confidence level in a counting time of less than 5 minutes. Table 8 provides the frequency or probability for observation of "n" or more counts from a wipe source of 20 dis/minute.

The standard presents examples of instruments and monitoring methods that will ensure compliance with surface contamination guidelines.

LIMITS

The standard was limited in that the following subjects were beyond its scope:

- Control of radioactivity that is dispersed in material, or both on the surface and dispersed in material;
- Surface radioactivity from radionuclides detectable only by beta particles with $E_{max} \leq 150$ keV;
- Radioactivity in soils; and,
- Radioactivity on clothing and persons.

The reviews by the ANSI N13 Committee and HPSSC resulted in the following objections:

1. The limit of 300 dis/min-100 cm^2 for most alpha emitters was too low to measure practically.
2. There was an objection to placing low limits on natural Uranium and Thorium than in previous versions, 200 removable and 1000 fixed, rather than 1000 removable

and 5000 fixed. It was also stated that there was no convincing evidence that uranium should not be included in the least restrictive category. "This standard, if adopted, will severely impact operations at many uranium handling facilities inferring, without evidence to substantiate such, that past operations have been health threatening."

3. The Appendix should provide a more thorough discussion to support the technical basis for the values in the draft standard.

4. There is no explicit distinction between fixed and loose radioactive contamination.

These objections were representative of three major classes of concerns expressed to the Working Group over the history of the standard - the ability to measure the limits, the assignment of radionuclides to specific groups, and the magnitude of the actual values of the limits, including the rationale for choosing the values. These points will be discussed in succession.

1. Some individuals believe that the limit of 300 dis/min-100 cm^2 for total alpha surface radioactivity was too low to measure. This conclusion was based on an extensive evaluation of survey instruments for alpha radiation by the Los Alamos National Laboratory and Pacific Northwest Laboratory (Olsher 1986). All models tested were zinc sulfide scintillation detectors.

One of the evaluation procedures was a survey of a 4 by 4-foot masonite surface divided into a grid of thirty-six 8 by 8 inch squares. Nine spots of alpha activity were painted on the surface with a Th-232-based paint. Alpha activity levels ranged from 64 dis/min to 672 dis/min. There were several different groups of participating survey monitors, including one group with no previous experience and one group consisting of experienced survey technicians. Participants used monitors with an audio output (one click for each count). They first estimated the background, and then very slowly scanned the surface of the grid listening for an increased "click" rate. Monitors were instructed to locate all of the hot spots and measure their emission rate in counts/min. They were given 10 minutes to monitor a surface. Based on the data a statistical analysis was used to determine the alpha source activity for a 50% detection frequency.

The best detection record was achieved by participants with no previous survey experience. They took a long time to monitor each surface, much longer than experienced personnel, and much longer than the 10 minutes allotted. The alpha source activity for a 50 percent detection frequency depended on the detector used. The values were as follows:

	Activity dpm	Standard Deviation dpm
Ludlum 3	305	23
Eberline ESP-1	376	43
Bicron Analyst	478	50
AN/PDR 60	301	33
AN/PDR 56F	1149	294

The probes used with the Ludlum 3, the Eberline ESP-1, and the Eberline PAC 4G have the following characteristics:

METER	EBERLINE ESP-1 AC-3-7 SCINT	EBERLINE PAC 4G AC-21 GAS PROP	LUDLUM 3 43-5 SCINT
Active area (cm^2)	50	50	50
Window (mg/cm^2)	0.5	0.85	1
Probe dimensions (cm)	14.4 X 5.08	18.5 X 3.3	19.7 X 5.7

The Ludlum AC-21 gas proportional probe used with the PAC 4G alpha monitor is included for reference. All probe backgrounds were about 1 c/min.

The ANSI N13.12 (1988) presents calculations of the expected response of a survey instrument to surface radioactivity over which it passes. The response is preferably in terms of aural clicks, although individual needle deflections can sometimes be detected. For a point source, the analysis is in terms of the actual time taken to pass over the source and the efficiency of detection. The time for a detector of length 14 cm to pass over a spot at the recommended speed of 10 cm/sec is 1.4 sec. A surface activity of 300/min and 30 percent detection efficiency would give an expected count of something less than 2.1 counts during traversal by the detector along its axis. The Poisson probability of observing no counts during the traverse would be $e^{-2.1} \times 2.1^0/0! = 0.122$ and the probability of observing 1 or more counts is 1-0.122 = 0.88. Thus the 300 dis/min spot of contamination seems is detectable, if borderline, and this is borne out by the survey results quoted above (Shapiro 1987).

2. Objections To The Uranium And Thorium Limits

The U.S. Atomic Energy Commission established a limit on the intake by inhalation of airborne uranium ore dust in 1960. In the following 20 years, it was learned that thorium-230, one of the long lived decay products that accompanies natural uranium, had a biological half life in the lungs of 1 year, in contrast to the 120 day biological half life for uranium previously assumed to also hold for thorium. To use this finding would have required a reduction in the permitted airborne concentration of uranium ore dust. However, according to McGuire (1983) there is a mitigating factor associated with uranium ore dust particle size. "...the uranium ore dust in uranium mills was found to occur with very large particle sizes (10-micron activity median aerodynamic diameter, AMAD)...The two effects are of about the same magnitude but in opposing direction. Thus the present uranium ore dust intake limit in NRC regulations should provide a level of protection consistent with that provided for other airborne radioactive materials."

Does the large median aerodynamic diameter in mills for uranium dust extend to uranium residual surface radioactivity in other locations? We don't know. However, the low specific activity of uranium has also been invoked by the IAEA (1963) as reducing the perceived hazard of surface radioactivity - "The specific activity of a radionuclide affects the probability that a given quantity of the radioactivity may enter the body and affects its subsequent behavior in the body...As the specific activity is an important inherent property of a radionuclide, it cannot be ignored in making a classification." IAEA ranked uranium 236 in toxicity among 263 radionuclides, in the same grouping as H-3 and I-129. Pu-239 was ranked 4, I-131 was 41. Wrixon (1979) divided radionuclides into two categories, the most hazardous and all the rest. He used a resuspension factor of 5×10^{-5} m^{-1} for all radionuclides except those with low specific activity. "There is some evidence that resuspension factors for low specific activity materials, such as natural uranium, are at least ten times lower than those found for plutonium...there is considerable merit in using a different resuspension factor for low specific activity materials, since such materials are frequently handled in the absence of other radionuclides...where resuspension is the main source of airborne contamination, the mass of material that would have to be resuspended is relevant."

Duggan (1972) noted that less restrictive toxicity classifications for Th-nat and U-nat given by IAEA were based on acute exposures with a maximum inhaled mass of 10 mg and that industrial exposures (presumably a more diffuse distribution of activity) will reach much higher levels since they occur continually over the year - "It is therefore concluded that in any toxicity classification of the radionuclides, which is to be applied in situations where continual exposure in industrial conditions is envisaged, both Th-nat and U-nat should be in one of the more toxic categories and not (as in the IAEA classification) in the group of lowest toxicity.

In considering all these factors, and the low limits for airborne concentrations of uranium, the working group felt that it could not retain the traditional lowest toxicity classification for natural uranium, but it also felt that it was being too conservative to classify it with the most toxic materials. For this reason, a separate category was established for the very low specific activity radionuclides, intermediate between the lowest and highest toxicity groups.

3. Analysis Of The Hazard From Surface Radioactivity

Early in its deliberations, the working group decided that a workable standard had to be a performance standard - that it was not possible to present a rigorous technical analysis to develop limits in terms of risk. In the control of nonradioactive materials, monitoring for surface contamination is often bypassed completely in favor of limiting control to air monitoring to determine compliance with prescribed limits for airborne levels.

Thus, the trial use of the standard over 15 years, including repeated critical reviews that accompanied successive balloting by the HPSSC, has served to validate its status as a performance standard. The limits and measurement methods appear to be practicable, and most critical comments require only minor changes. Methods of calculating the hazard from surface radioactivity are given in the literature, however, and it is of interest to apply these methods to hazard evaluations at the limits given here or to compare recommended limits. This should provide an indication of the magnitude of control achieved.

The standard presents an equation for the concentration of activity in the air as a function of the surface radioactivity. An equilibrium situation is assumed where the activity is removed by ventilation at the same rate at which it becomes airborne. The equation is:

$$\text{Concentration} = \frac{\text{Surface activity X Area X fractional removal rate}}{\text{Room volume X rate of air changes}}$$

$$\text{dis/min-m}^3 = \frac{\text{dis/min-m}^2 \text{ X m}^2 \text{ X fraction/h}}{\text{m}^3 \text{ X air changes/h}}$$

The standard gives an example using Area/Volume = 0.44 m^{-1}; 5 air changes/hr; fractional removal rate of 10^{-5}/h. At the guide level for plutonium of 30,000/min-m^2, the equilibrium activity works out to 1.2 X 10^{-14} uci/ml, which is 20 percent of the limit for Pu in the environment given in 10 CFR 20.

Another approach that is often used is in terms of the Resuspension factor. This is the ratio of the air concentration to the surface concentration of activity. It may be derived from the equation above as:

$$\text{Resuspension} = \frac{\text{dis/min-m}^3}{\text{dis/min-m}^2} = \frac{\text{m}^2 \text{ X fraction/h}}{\text{m}^3 \text{ X air changes/h}}$$

The equivalent resuspension factor for the example above is:

$$0.44/\text{m X } 10^{-5}/\text{h} - 5 = 0.9 \text{ X } 10^{-6}$$

It is obvious that the evaluation of airborne activity, whether through the use of a rate of resuspension or as a resuspension factor, is strongly dependent on values which are difficult to assign in a general case. Healy (1971) has proposed resuspension rates which are "reasonably, but not overly, conservative" varying from 5 X 10^{-3}/h for vigorous activity, including cleaning or children at active play to 10^{-6}/h for "quiet, no movement." Tabulated values of resuspension factors have varied from 10^{-3}/m to 10^{-8}/m.

In one example in the standard, it is assumed that a person might ingest all the contamination from 10 cm^2 of surface each day, an assumption first used by Dunster (1962) and quoted widely since. The consequences of this assumption are derived by referring the calculated intake to the Annual Limit on Intake for the radionuclide considered.

The ALI given in ICRP 26 for Radium-226 is 200,000 Bq and this results in a whole-body dose commitment of 5000 mrem. For a daily intake at the alpha limit of 300 dpm/100cm^2 (amounting to 0.5 Bq/day), the committed dose from a year's intake is 4.6 mrem. Because of the long half-lives, the committed doses are not accumulated until after many years, so the actual annual dose is much less.

Various other modeling exercises have been done. All come up with doses below limits set by professional and regulatory agencies at the surface radioactivity levels given in the standard.

4. On The Distinction Between Fixed And Loose Radioactive Contamination

Healy (1971) did not distinguish between fixed and loose contamination - "In view of the possibility of abrasive or similar actions on items during repair or modification, it is believed that these limits should apply to the total contamination and not just to the "loose" contamination unless other information rules out such actions on the item."

Wrixon et al. (1979) made note of the problem of distinguishing between fixed and loose radioactivity - "By definition, resuspension and subsequent inhalation of surface contamination can occur only with loose activity. A clear distinction between loose and fixed contamination cannot be readily made however. What appears at one time to be firmly fixed may, through movement and other physical processes, become loose. Furthermore, the resuspension factors used in the calculations here are derived from a consideration of the total surface activity. The activity removed by wiping the surfaces involved in the experiments was of the order of 10% of the total activity; in these cases also, the Derived Levels (DLs) are normally appropriate for the direct measurement of contamination. If, however, the contamination clearly is firmly fixed, higher levels than the DLs may be permitted."

"It is recommended therefore, that monitoring be carried out by direct measurement. Where this is not feasible, as for example, in the presence of high levels of gamma radiation, the normal methods of wiping should be used and the assumption made that only 10 percent of the surface contamination has been removed."

After considering the arguments for relying solely on direct measuring in monitoring for surface radioactivity, the Working Group felt it prudent to retain the use of both direct and indirect measurements. Depending on the radionuclide and the monitoring circumstances, the wipe test could be the method of choice for finding surface radioactivity. In any event, it served as a check on the direct measurement to provide an additional control for compliance with the standard before release.

CONCLUSION

The guidelines in the proposed ANSI N13.12 are not based up a risk or safety analysis. The relationship between dose and surface radioactivity is very tenuous. Some calculations and analysis are presented in an appendix to the standard and thus are not part of the standard itself.

The proposed ANSI N13.12 standard was developed as a consensus performance standard and has been a de facto standard for over a decade. The lowest limit that by consensus are achievable has evolved by a continuous evaluation of the practicability of decontaminating equipment to prescribed limits and on detecting those limits. The guide concentrations in the proposed ANSI N13.12 were developed as a performance standard, tested through use and consensus.

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TABLE 1
Surface Contamination Limits:
Abbreviated Table From Proposed ANSI N328 (1974)

Nuclide	dpm/100 cm^2 Total	Removable
Group 1: Nuclides for which the MPC for air is 2×10^{-13} Ci/m^3 or less or for which the MPC for water is 2×10^{-7} Ci/m^3 or less; includes Pu-239, Ra-226, Pb-210, I-125, I-129, etc.	100	20
Group 2: Those nuclides not in Group 1 for which the MPC for air is 1×10^{-12} Ci/m^3 or less or for which the MPC for water is 1×10^{-6} or less; includes Po-210; Sr-90; Th-232; U-232, etc.	1,000	200
Group 3: Those nuclides not in Group 1 or Group 2.	5,000	1,000

TABLE 2
Surface Contamination Limits:
Abbreviated Table From Regulatory Guide 1.86 (1974)

Nuclide	dpm/100 cm^2 Average Total	Maximum Total	Removable
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100	300	20
Th-Natural, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000	3,000	200
U-Natural, U-235, U-238, and associated decay products	5,000	15,000	1,000
Beta-gamma emitters except Sr-90 and others noted	5,000 ßγ	15,000 ßγ	1,000 -γ

TABLE 3
Surface Contamination Limits:
Abbreviated Table From Draft ANSI N13.12 (1978)

Nuclide	dpm/100 cm^2 Total	Removable
Group 1: nuclides for which the MPC for air is 2×10^{-13} Ci/m^3 or less or for which the nonoccupational MPC for water is 2×10^{-7} Ci/m^3 or less; includes Pu-239, Ra-226, Pb-210, I-125, I-129, etc.	Nondetectable	20
Group 2: those nuclides not in Group 1 for which the MPC for air is 1×10^{-12} Ci/m^3 or less or for which the MPC for water is 1×10^{-6} Ci/m^3 or less; includes Po-210; Sr-90; Th-232; U-232, etc.	2,000 Nondetectable $\beta\gamma$	200
Group 3: those nuclides not in Group 1 or Group 2	5,000	1,000

TABLE 4
Surface Contamination Limits:
Abbreviated Table From Proposed ANSI N13.12 (1981)

Group	Description	dpm/100 cm^2 Total	Removable
1	Alpha emitters except Nat-U and Nat-Th	300	20
2	Sr-90, I-125, I-129, I-131, Ra-228	5,000	200
3	Beta emitters not in Group 2 with $E_{max} > 150$ keV; Nat-U and Nat-Th	5,000	1,000

TABLE 5
Surface Contamination Limits:
Abbreviation of Table From Proposed ANSI N13.12 (1983)

Group	Description	dpm/100 cm^2 Total	Removable
1	Alpha emitters except Nat-U and Nat-Th, depleted uranium, U-238 and Th-232	300	20
2	Sr-90, I-125, I-129, I-131, Ra-228	5,000	200
3	All radionuclides not in groups 1 or 2 except beta emitters with E_{max} < 150 keV	5,000	1,000

TABLE 6
Surface Contamination Limits:
Table From Proposed ANSI N13.12 (1988)[a]

Group	Description	ACTIVITY GUIDE[b] (dpm/100 cm^2) Removable	(Fixed plus removable)
1	All alpha emitters except those with extremely low specific activity and their associated decay products as listed in Group 4; Pb-210, Rac-228	20	300
2	Sr-90, I-125, I-129 I-131[d]	200	5000
3	All beta and gamma emitters not specified in Groups 1, 2, and 4 except pure beta emitters with $E_{max} \geq 150$ KeVe	1000	5000
4	Uranium (natural, depleted, enriched), Th-nat[f]	200	1000

a. A rationale for these surface activity guides is presented in Appendix B. Where both alpha and beta-gamma emitting radionuclides exist, the limits tablished for alpha and beta-gamma emitting nuclides shall apply independently.

b. The levels may be averaged over one square meter provided the maximum surface activity in any area of 100 cm^2 is less than three times the guide values. For purposed of averaging, any square meter of surface shall be considered to be above the activity guide G if: (1) from measurements of a representative number n of sections it is determined that $1/n \, \Sigma_n \, S_i > G$, where S_i is the dis/min-100 cm^2 determined from measurement of section i; or (2) it is determined that the sum of the activity of all isolated spots or particles in any 100 cm^2 area exceeds 3G.

   For purposes of this standard, the disintegration rate refers to those disintegrations which result in the emission of alpha particles, beta particles or electrons with $E_{max} \geq 150$ KeV, or photons of energy ≥ 20 KeV, provided these particles are responsible for more than 80% of the dose.

c. Pb-210 is included due to an alpha emitter, Po-210, in its decay chain and Ra-228 is included due to an alpha emitter, Th-228, in its decay chain.

d. These are the radionuclides undergoing beta or electron capture decay that present the greatest hazards as surface radioactivity.

TABLE 6 (con't)

e. The pure beta emitters with maximum energy less than 150 KeV are excluded because detection by direct methods is not practical and they must be treated on a case-by-case basis. However, radionuclides that are detectable by direct mesurement with appropriate instrumentation through emission of low-energy X and gamma rays (as in electron capture) or through the presence of short-lived decay products are included in this category.

f. U-nat and Th-nat include gross alpha disintegration rates of natural uranium, depleted uranium, uranium enriched to less than 10% U-235, Th-232, and their decay products.

TABLE 7
FREQUENCY FOR PROBABILITY FOR OBSERVATION WITH A SURVEY INSTRUMENT OF "n" OR MORE COUNTS[a] FROM A POINT SOURCE OF 300 DIS/MIN, ALPHA

E[c] (counts per disintegration)	t[b] (seconds counting time)	%		
		n = 1	n = 2	n = 3
0.5	1.67	99	92	79
	1.0	92	72	46
	0.67	81	50	--
	0.5	72	36	--
	0.33	57	--	--
0.4	1.67	97	86	65
	1.0	87	60	33
	0.67	74	39	--
	0.5	64	--	--
	0.33	49	--	--
0.3	1.67	92	72	46
	1.0	78	45	--
	0.67	64	--	--
	0.5	53	--	--
	0.33	40	--	--
0.2	1.67	82	51	24
	1.0	67	27	--
	0.67	49	--	--
	0.5	--	--	--
	0.33	--	--	--

a. Background is 1 (cpm).

b. The survey speed of the detector is v = d/t(cm/s) where t is the counting time and d is the width of the detector.

c. The mean counts/disintegration with the source under the window of the stationary detector.

TABLE 8
FREQUENCY FOR PROBABILITY FOR OBSERVATION OF "n" OR MORE COUNTS[a] FROM A WIPE SOURCE OF 20 DIS/MIN, ALPHA

E[c] (counts per disintegration)	t[b] (seconds counting time)	%			
		n = 1	n = 2	n = 3	n = 4
0.1	0.5	78	--	--	--
	1.0	95	80	58	--
	1.5	99	94	83	66
	2.0	100	98	94	85
0.2	0.5	92	71	--	--
	1.0	99	96	96	74
	1.5	100	100	98	94
	2.0	100	100	100	99
0.3	0.5	97	86	68	--
	1.0	100	99	97	92
	1.5	100	100	100	99
	2.0	100	100	100	100
0.4	0.5	99	94	83	66
	1.0	100	100	99	98
	1.5	100	100	100	100
	2.0	100	100	100	100
0.5	0.5	100	97	91	80
	1.0	100	100	100	100
	1.5	100	100	100	100
	2.0	100	100	100	100

a. Background used is 1 cpm.

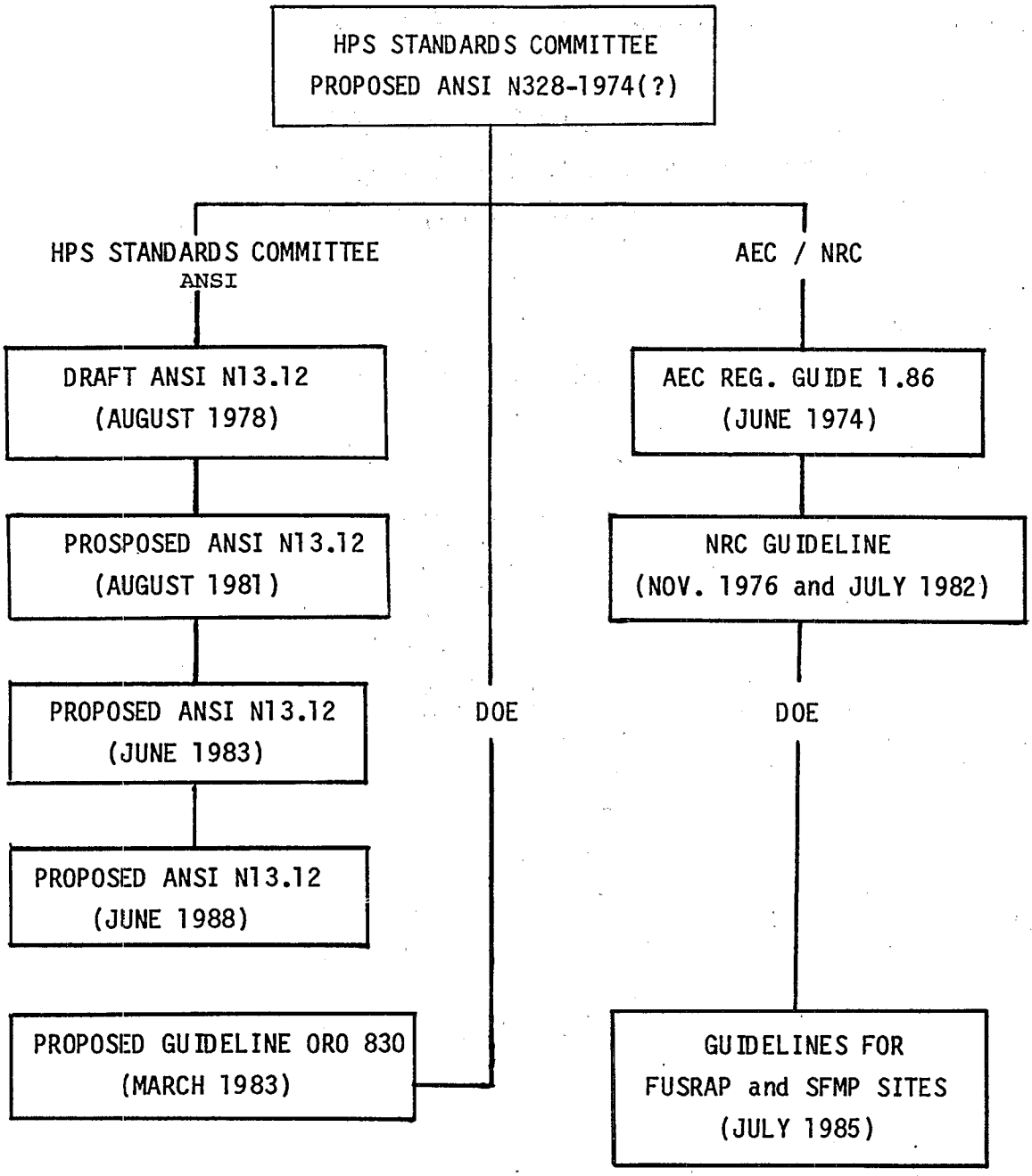

Figure 1. Interrelation of Surface Contamination Guidelines.

EPA's Proposed Environmental Standards for Low-Level Radioactive Waste Disposal and Criteria for Below Regulatory Concern

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ABSTRACT

The Environmental Protection Agency has developed generally applicable environmental standards for land disposal of low-level radioactive waste. The elements of the proposed standards will include: (a) exposure limits for pre-disposal management and storage operations; (b) criteria for other agencies to follow in specifying wastes that are Below Regulatory Concern (BRC); (c) post-disposal exposure limits; (d) ground-water protection requirements; and (e) qualitative implementation requirements.

To support the concept of BRC, the Agency has developed technical information, cost data and a risk assessment methodology for analyzing promising candidate waste streams. The BRC criteria are based on general population health risks, maximum annual exposures to critical population groups, and the costs now associated with the regulation of these wastes.

The regulatory package for these standards is presently under interagency review. Promulgation of the proposed standards is expected in 1990.

INTRODUCTION

In August 1983, the Environmental Protection Agency (EPA) published an Advanced Notice of Proposed Rulemaking (ANPRM) [1], stating the Agency's intention to develop generally applicable environmental standards for the land disposal of low-level radioactive waste (LLW). The intent is that these standards (40 CFR 193) must be met by facilities that dispose of LLW, whether the facilities are licensed and regulated by the Nuclear Regulatory Commission (NRC) or their Agreement States, or are owned and operated by the Department of Energy (DOE).

The EPA LLW Standard is intended to cover disposal of all AEA materials not covered by other EPA Standards, i.e., all radioactive waste that is not high-level and transuranic radioactive waste or spent nuclear fuel, as defined in 40 CFR Part 191, or, uranium or thorium mill tailings subject to 40 CFR Part 192. The Standard will have the following elements:

(a) Low-level waste pre-disposal management and storage. This would include limits on radiation exposure to individuals during processing, management, and storage of LLW.

(b) Definition of radiation exposures related to low-level radioactive waste disposal that are sufficiently small that they do not need to be regulated regarding their radiation hazard (i.e., a level "below regulatory concern").

(c) Limits on radiation exposure to individuals after the disposal site is closed.

(d) Ground-water protection requirements for both pre- and post-disposal phases.

(e) Qualitative implementation requirements.

STANDARDS RATIONALE

Individual Radiation Exposure Limits During Management and Storage (Pre-Disposal)

This element would limit annual effective whole body exposure from all environmental pathways to any member of the public from facilities which process, manage, or store LLW. This would include the operation phase of regulated LLW disposal facilities, i.e., while they are receiving and emplacing waste; and "away from generator" LLW management, processing and storage facilities.

The Office of Radiation Program's analyses indicates a standard around 25 mrem/yr from all pathways would be consistent with the technology and other similar standards.

"Below Regulatory Concern" Criteria

Criteria are being proposed for identifying LLW with sufficiently low levels of radioactivity to qualify as "Below Regulatory Concern" (BRC). Any waste meeting these criteria could be disposed of as a non-radioactive waste. However, if it had Resource Conservation and Recovery Act (RCRA) hazardous characteristics, it would have to be disposed of in compliance with RCRA regulations. The EPA would not be involved in identifying or selecting specific LLW types which qualify as BRC wastes, the NRC, States and DOE would implement the use of our criteria for determining which wastes would qualify for disposal by less restrictive means.

In arriving at a proposed BRC level, EPA carefully weighed and considered many factors. The Office's analyses has indicated that a standard around 4 mrem/yr would provide protection of the public and the environment.

Individual Radiation Exposure Limits for Post-Disposal

Our standard will establish limits on exposure through all pathways to members of the public from the land disposal of LLW. EPA's post-disposal limit would apply to any DOE or NRC/State-licensed LLW land disposal method or facility constructed after the effective date of the rule and apply to existing disposal facilities within a certain time frame.

A persuasive reason would be needed to significantly depart from a 25 mrem/yr level. EPA's technical analysis has not revealed any such reason so far.

Ground-Water Protection

The protection of the Nation's groundwaters is of major importance in EPA and such a consideration is particularly appropriate in land-based waste disposal standards.

Two sets of ground-water protection requirements will be proposed and public comments solicited. In both proposals Class I groundwaters require the highest levels of protection and represents those that are highly vulnerable to contamination and serve as irreplaceable sources of drinking water for large populations. It is appropriate to give these groundwaters the highest level of protection, i.e., non-degradation. The two proposals differ only with respect to the protection levels for Class II groundwaters which represent all non-Class I present or potential sources of drinking water. The first proposal would protect Class II groundwaters from high yield aquifers (which are or could be a community water supply) to an annual effective dose equivalent of 4 mrem, while Class II groundwaters from low yield aquifers (which generally could not provide a community water supply) would be protected as a part of the 25 mrem/yr all pathways pre- and post-disposal performance standards. The second proposal would protect all Class II groundwaters, which is by far the largest category of groundwaters, to an annual effective dose equivalent of 4 mrem. This level is comparable to the 4 mrem/yr Maximum Contaminant Level (MCL) for manmade beta particle and photon radioactivity established for public water supplies by EPA's drinking water standards under the Safe Drinking Water Act [2].

Finally, both proposals recommend the same levels of protection for Class III groundwaters. Class III A groundwaters are protected to the level applicable to the highest class of groundwater to which it is interconnected. Class III B groundwaters have a low degree of interconnection with other classes of groundwater and would be protected as a part of the 25 mrem/yr all pathways pre- and post-disposal performance standards.

Qualitative Requirements

Qualitative requirements are being proposed which would make clear the context and assumptions within which we expect the Standard to be implemented.

These requirements would address areas not appropriate for quantitative requirements and compensate for the uncertainties that necessarily accompany plans to isolate radioactive

wastes from the environment for a long time. They would include: (a) limiting the dependence on active institutional controls (such as guarding, maintenance or cleanup of releases) after disposal to no more than 100 years, (b) providing passive institutional measures (such as permanent markers, records or archives or government ownership) which should reduce the chance of inadvertent human intrusion beyond the active institutional control period; (c) requiring monitoring during disposal and post-disposal phases which should be done with techniques that would not jeopardize the isolation of the wastes; and (d) suggesting site location away from areas containing materials not widely available from other sources (such as minerals, fuels and groundwaters).

BRC RATIONALE

Philosophical Approach

We believe that when some LLW streams contain sufficiently small concentrations of radioactivity, there is no reason, from a public health point of view, not to dispose of these wastes as we would any "non-radioactive" trash. However, this is not to say that there is no remaining risk or that any form of waste disposal is risk free. EPA recognizes that this remaining risk should be estimated, and adequate assurance must be given that it is not unreasonable in light of the benefits of deregulating it.

It should be noted that EPA's concept of BRC is not one of a de minimis level. BRC is based on a careful analysis of specific sources of exposures, e.g., low-level radioactive waste disposal, and the methods of exposure control. A BRC decision is a conclusion, that relative to a specific practice, certain radiological impacts will be small and not worth the effort that would be necessary to further reduce them. The concept of a de minimis level does not consider specific practices, costs or facility locations, but refers to a level of negligible risk from all sources of radiation exposure or any other potential insult. If a generic de minimis risk level were determined, it would be at or below any BRC level.

Concepts for Establishing BRC Levels

We believe there are two essential elements when deciding not to regulate a beneficial practice which can cause radiation exposure. The first is that the deleterious impact of the practice on health in the exposed population, taken as a whole, is small enough that the effort and expense of regulation is not warranted. The second is that the risk to any person is small compared with other risks in society.

In developing our BRC proposal for low-level waste an initial premise was that since this would be a criterion for not regulating certain waste streams, and that wastes that met the criteria would not receive any monitoring or follow-up, there would be no long-term confirmation of the results. This implies that the BRC level should be well below the limit for regulated disposal. Additionally, as we see it, the concept of "As Low As Reasonably Achievable" (ALARA) is indeed related to BRC. Such a regulatory cutoff may be appropriately viewed as a floor to ALARA. The proper perspective, we believe, is that there is a standard or upper limit (the level for regulated disposal) below which one practices ALARA, and then at some lower level (BRC) there could be

a regulatory cut-off where we say that for this practice it isn't worth the effort to go any lower. This should allow regulatory resources to be expended more effectively.

Basic Criteria

There are several necessary considerations in determining a regulatory cutoff. One of the most important of these is economics, i.e., the costs of implementing control. There are two important consequences of this: First, there may be no single number applicable to all practices as the costs of control may vary with practice. Therefore, there may be a separate and distinct number associated with each activity or "practice." Second, as we previously indicated, such a cutoff by the nature of its determination, serves as a floor to "As Low As Reasonably Achievable" (ALARA), but only for the specific practice.

We believe that a BRC decision consideration should at least:

1. Exempt a relatively small collective dose which does not significantly change with individual dose in the area of the BRC decision. We have no magic number for this collective dose, but its size should be considered relative to the total collective dose if the whole of the practice were unregulated.

2. Represent an individual dose that is well below the overall regulatory limit for the practice. This is so it will represent a small individual risk and can be readily differentiated from the regulatory limit, which is presumed to represent an acceptable risk.

3. Be formulated so that it does not increase other environmental impacts.

4. Be compatible with legal authorities and other control actions dealing with nonradioactive pollutants.

5. Be able to be practically implemented using available management systems, analytical techniques, and instruments.

6. Result in the possibility of some resource savings for the regulator and the practice. However, if it holds little hope for ultimately saving resources, it would negate one of our prime reasons for considering a BRC in the first place (i.e., allocation of resources to the more significant risks).

7. Be supported by an analysis that provides a reasonable and sufficient basis for the decision-makers to arrive at a judgment.

Risk Assessment Methodology and Health Impacts

To set a BRC level that would provide adequate public health and environmental protection, a methodology was developed for assessing health impacts, i.e., the cumulative population health effects and critical population group (CPG) dose exposure.

To estimate the possible doses and economic impacts resulting from potential BRC deregulation, EPA modeled several scenarios of possible waste streams,disposal methods, and

various demographic and hydrogeologic/climatic settings. Surrogate types of LLW were chosen to represent a wide variety of waste generators, such as power reactors, uranium fuel fabrication and processing facilities, industrial facilities, medical facilities [3]. The BRC scenarios included a variety of disposal methods, i.e., municipal sanitary landfills, dumps, on-site landfills, and incineration methods situated in rural, suburban, and urban demographic settings.

In determining radiation doses for the CPG who might collect the wastes, on-site disposal facility workers doing routine incineration and disposal operations, reclaimers, inadvertent intruders, and nearby residents exposed to water, food, and inhalation pathways were included. Health effects to the general population over 10,000 years were calculated. Individual radiation doses were calculated as a committed annual effective radiation dose equivalent for 10,000 years to an individual in the CPG. Estimated lifetime risks were also calculated for the CPG [4].

Cost Benefit Analysis

EPA also performed a national economic assessment of potential BRC waste streams. A cost-benefit evaluation was done for a range of alternative exposure levels from 0.1 to 15 mrem/yr. All of the alternative non-zero BRC exposure levels could reduce the volume of regulated LLW. Volumes of regulated waste could be reduced by up to 43 percent, with attendant cost-savings ranging up to over $700 million over a 20-year period, except at the very low levels of exposure where it would require considerable expense to regulate those materials presently not being regulated [4,5].

EPA's Approach to the Proposed Criteria

Foremost in our approach was protection of the public and the environment. Our approach was to develop an exposure level with assurance of no undue risk.

EPA believes that at the BRC level, the health risk should be very small when compared to other risks we encounter in our daily lives. The Agency considered: (a) the risk from natural background radiation, which is on the order of 3×10^{-3} to 10^{-2} lifetime; and (b) the lifetime risk of developing fatal cancer, which is about 2×10^{-1}.

Another consideration was that if a person were exposed to several deregulated waste streams, the total risk should still be small. Our health impact analysis, even using multiple waste streams, indicated lifetime risks for many of the disposal scenarios to be less than 1×10^{-4}.

Based on the technical and economic evaluation, EPA considered several specific regulatory options for BRC all of which represented levels of protection more stringent than for proposed limits on regulated LLW disposal. These options ranged upward from the position that there would be no BRC criteria allowed, in which case the regulated disposal of all wastes would be required as long as any radioactivity remained.

The other options considered included criteria for the less restrictive disposal of BRC with limits of 0.1 mrem/yr, 1 mrem/yr, 4 mrem/yr and 15 mrem/yr to the individual which results in a lifetime risk of 2.8×10^{-6}, 2.8×10^{-5}, 1.1×10^{-4} and 4.2×10^{-4}, respectively.

Another regulatory option was to accept the current practice, wherein EPA would either stop its BRC effort or endorse the existing practice of case-by-case deregulations.

In considering a BRC level, we also took note of other regulated risk levels used by other government programs. Of special note was the 4 mrem/yr dose level ($1.1x10^{-4}$ lifetime risk) set for man-made radionuclides in the National Interim Drinking Water Standards (2).

Another consideration was the cumulative population health effect. In our BRC range (0.1 to 15 mrem/yr) the additional health effects would be of the order of less than 0.001 to 0.05 per year over the 10,000 years.

All of these considerations and evaluation together with our risk analysis indicate that many wastes could be disposed of without consideration of their very low levels of radioactivity. As a result, we are proposing a BRC value of 4 mrem/yr (a lifetime risk of $1.1x10^{-4}$). We believe this provides reasonable assurance of protecting the public and the environment with minimal incremental risks.

Implementation Guidelines

There are several basic factors EPA considers necessary for an agency making a specific waste stream deregulatory decision.

An evaluation of the collective dose, and thus the risk to the population, is a necessary part of any cost-benefit evaluations. It must be estimated to assure there is no significant impact on public health. It will also serve as an indication of, and thus a measure to deter the practice of, waste dilution to reach the BRC individual exposure limits. Regardless of whether collective dose is a specific parameter in any BRC limits, the total potential health impact of an exemption needs to be considered in some way as a part of any decision for deregulation.

It is also important to be able to characterize with reasonable certainty the waste streams' physical, chemical, and radiological characteristics.

The waste should have negligible potential recycle value so it will not be attractive to scavengers who might otherwise take the waste and use it.

It is important that sufficient recordkeeping and reporting requirements be included to provide the regulatory agency with adequate information to understand where and how much BRC waste is being disposed of. Records containing this information will be needed to document how the BRC concept is working in actual practice and will serve as a "final accounting" for the BRC wastes before regulatory control is irreversibly lost.

Consistency With International Policy

The Agency also reviewed the guidance for similar exemptions being developed by other countries and international organizations. The Canadian Atomic Energy Control Board [6] and the United Kingdom's National Radiological Protection Board [7] have issued documents proposing a 5 mrem/yr dose criterion to members of the public to be used for case-by-case analysis of applications for license exemptions for radioactive waste disposal.

The International Atomic Energy Agency (IAEA) has considered a de minimis dose of 1 mrem/yr to the average individual in the CPG for ocean dumping and for quantities of solid radioactive waste for uncontrolled disposal by incineration and landfill [8,9]. The International Commission on Radiological Protection has also adopted a radiation principle of an annual individual dose exemption criterion of 1 mrem [10]. These latter de minimis recommendations differ from BRC, in that they are value judgments of negligible risk and make no attempt to consider the cost of regulation.

In 1988, the IAEA issued its recommendations on "Principles for the Exemption of Radiation Source and Practices from Regulatory Control." They concluded that individual doses of about 1 mrem/yr from each exempted source, leading to a total dose from all such sources of a few mrem/yr, were reasonable, but only if the societal impact was sufficiently low. In this regard, they suggested that 100 man-rem from the entire practice would qualify it for exemption without further analysis.

CONCLUSION

We are now in the final stage of proposing the LLW waste standards. We have prepared regulatory support documents which will be available when the proposed standard is published in the Federal Register.

The EPA Low-Level Radioactive Waste Management program staff believes the Standards covering the above described areas would provide adequate protection to members of the general public with a reasonable balance of risks and costs.

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EPRI Discussion Paper on BRC and De Minimis Concepts

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ABSTRACT

The purpose of this discussion paper is to examine the definitions, relationships and characteristics of the terms: de minimis, below regulatory concern (BRC) and generic BRC. In the past few years these terms have been used in various contexts such as: disposal of very low-level wastes, dose levels to members of the public, and the unrestricted release or use of slightly radioactive materials. While all of the terms have been taken to imply radiation risk levels of little significance, the use of the terms has not been wholly consistent and there appear to be multiple and sometimes conflicting interpretations of the terms. The use of these terms is explored herein. In addition to this discussion paper, the results of the Electric Power Research Institute (EPRI) $2M research program to establish the technical basis for petitioning the NRC to exempt very low-level nuclear reactor wastes from LLW disposal facilities are also summarized in this paper.

INTRODUCTION

The generally accepted uses of these terms, as discussed in detail below, are:

de minimis: a low exposure level based on the corresponding risk which has been determined to be neglible or trivial based solely on a comparison with other generally accepted risks.

activity-specific BRC: an exposure level set for a specific regulated activity based on an cost-risk reduction evaluation showing that further controls to reduce exposures beyond are not justified.

generic BRC: an exposure level set for a class of comparable activities based on a cost-risk reduction evaluation of a typical activity in that class showing that further controls to reduce exposures beyond this level are not justified.

DE MINIMIS

The term "de minimis" is an abbreviation of the phrase "de minimis con curat lex" which implies: the law is not concerned with trivialities. The initial use of the term in radiation protection matters was to identify the need for establishing or codifying a radiation exposure level (or risk) to a member of the public which would be considered trivial by the members of the public. The approaches suggested for the establishment of such a dose were generally on a comparative risk basis against other society risks readily accepted or against naturally occurring background radiation levels. From a regulatory point of view, any activity or practice which had projected doses at or below the de minimis dose would not require further regulation, including actions or controls to reduce the doses below the de minimis level. Thus, implicit in the de minimis definition is that a COST-BENEFIT evaluation would not be required at this dose level to determine if further dose reductions can be justified. Likewise, the definition indicates that the de minimis dose would be universally applicable to any activity or practice which had the potential to expose members of the public. Thus the important features of a de minimis exposure limit, identified from the definitions and uses, are that the dose level would apply to all activities and would be based on a comparative risk basis, rather than a COST-BENEFIT basis.

BRC

Subsequent to these initial discussions, the de minimis concept was supplanted with the concept of below regulatory concern (BRC). This new concept was intended to differentiate between de minimis, which implies the establishment of a trivial risk without the need for additional regulatory considerations, and BRC which would embody other regulatory considerations such as cost/benefit or evaluations.

Although this seems to be the basis for the introduction of the BRC concept, considerable confusion has since resulted in the use and application of the term. Much of the confusion may stem from the unfortunate choice of the words "below regulatory concern" which connote a degree of regulatory triviality rather than a dose level which represents a regulatory cost-effectiveness limit or objective. For example, BRC is often described using words such as: "minimal risk," "trivial risks from a regulatory standpoint," "insignificant risk level," all of which tend to emphasize the smallness of the risk as the basis for the dose level rather than the fact that further regulatory efforts to reduce the dose below the BRC dose level are not considered cost-effective on the basis of a cost-risk reduction evaluation. Examples of cost-effective limits are demonstrated by the 10CFR50, Appendix I dose objectives and the 40CFR190 dose limit both of which are cost-benefit based limits, but which contain no reference to BRC. Obviously, these cost effective limits represent a dividing line between doses which are above a cost-effective level and are therefore of regulatory concern and doses which are below regulatory concern for these activities. Doses at the limits are neither above regulatory concern nor are they below regulatory concern. Perhaps a better choice of words for a cost-effective objective or limit which would fall between the adequate protection limit and a de minimis level would be a Regulatory Cost-Effective Limit rather than BRC.

GENERIC BRC

Although there have been recent discussions regarding the establishment of a "generic BRC" dose or risk, the basis for establishing a generic BRC dose level is not clear. If generic BRC is intended to apply all regulated activities then it isn't clear that a generic BRC level can be established by a cost-risk reduction evaluation without doing an injustice to BRC levels which would be established on a source-specific basis. By their nature, cost risk reduction evaluations are performed on a source-specific and regulatory control-specific basis in order for the cost-effectiveness evaluation to be valid. If a generic cost-effectiveness evaluation is performed for all sources and all control measures then it is likely that a more restrictive dose level would be justified on the basis of the most cost-effective control measures applied to the easiest source to control. This would, in effect, invalidate the cost-effectiveness evaluations for some of the other sources.

A clear example of how a cost-benefit evaluation of too broad a class of unrelated activities could lead to an inappropriate generic BRC level is provided by EPA's evaluation of low level waste disposal. EPA lumped together both reactor generated and medical wastes even though only medical wastes contain significant quantities of carbon-14. As a result, the carbon-14 in medical wastes led to a significantly lower generic BRC level than would have resulted from an evaluation of reactor wastes alone. This shows that to be meaningful, a generic BRC level must be calculated only for a class of related activities which can be represented adequately as a typical member of that class.

Notwithstanding the potential for overly restrictive doses being applied to some activities or practices, a conservatively low generic BRC dose level may still be beneficial for licensees who would not choose to perform a source-specific cost-benefit evaluation which potentially could justify a higher dose level. The obvious benefit would be the elimination of the need for holding an additional rulemaking on a proposed activity-specific BRC dose level. However, there may be other ways to eliminate the need for source specific rulemaking, as discussed below under implementation. Even if a generic BRC dose level was established, there should be a higher dose level allowed as indicated by the results of the cost-benefit evaluation for those licensees that would choose to perform a cost-benefit evaluation for a proposed practice.

MULTIPLE SOURCES

To account for the potential for multiple sources or multiple exposures to a maximum exposed individual from more than one source it is important to identify the dose level for which the accounting is important. For example, the basic regulatory dose limit (proposed 100 mrem/yr) is fundamentally a risk-based limit and as such multiple sources must be taken into account to ensure that the limit is not exceeded for the maximum exposed individuals. Likewise, the de minimis dose level is also risk-based and therefore should take into account multiple exposures. It would be preferable to account for the potential of multiple source exposures at the de minimis level in the licensee's compliance implementation rather than by an arbitrary, and likely overly conservative, reduction in the de minimis dose level. Arbitrary factors of 5 to 10 have been suggested to account for multiple sources or exposures. However, in reality for most of the regulated activities the likelihood of overlap in the exposure of the maximum exposed individual from several activities is exceedingly small.

Because the BRC dose levels are cost-benefit and not risk-based, there is no need to account for multiple sources by reducing the BRC dose levels by some arbitrary factor unless there is a potential to exceed the basis regulatory dose limit. In fact further reductions in any cost-benefit BRC dose level invalidates the cost-benefit evaluation on which the dose level is based. And, as noted above for a de minimis dose, the potential for any significant overlap of exposures to the maximum exposed individual for more than one of the BRC activities is extremely remote. However, in the unusual case where there is a significant potential for multiple exposures, or where a BRC level is a significant fraction of the basic radiation limit, a more detailed evaluation of the potential for multiple exposures could be required.

IMPLEMENTATION

DeMinimis

It would be necessary to conduct a rulemaking to establish a de minimis dose or risk level. The rulemaking would focus on the basis for establishing that a given risk would be considered trivial by a member of the public. In addition to establishing a regulatory cut-off, the rulemaking could also establish the need for the NRC to evaluate and accept the dose computation models used to determine the expected doses from a proposed practice or activity. If NRC pre-approval of the dose compliance models is required, it would allow the NRC to account for the potential for overlap of exposures from multiple sources and also to account for recycle, re-use or use of licensed materials in indeterminate exposure pathways. Both overlap and indeterminate pathways could be accounted for by approving a dose limit for specific applications, which would be below the codified de minimis dose. As an alternative to this approach, the de minimis dose limit established in the regulations could be reduced by some arbitrary factor to account for multiple sources. It would still be necessary to account for indeterminate exposure pathways in the dose computation models.

BRC

To establish a BRC dose level for a source-specific practice, the NRC could follow the procedure defined by the NRC policy statement on BRC waste disposal wherein waste generators on a national scale would petition the NRC for a rulemaking to allow waste disposal of wastes by means other than in an NRC-licensed disposal facility as long as the doses were within the BRC level. However, in following this process, the phrase 'a few millirems' will not be taken as a pre-determined number suitable for a risk based limit. Rather, the phrase would be interpreted as implying that a cost-benefit evaluation is the appropriate process for setting the BRC limits, and the regulations would prescribe the need for a cost-benefit evaluation in accordance with the definition for a BRC dose level. Thus, consistent with the definition of BRC, the regulations should not specify any exposure limits but would require applicants to establish the appropriate exposure limits by a cost-benefit evaluation. It would also be beneficial for the NRC to publish general guidance on an acceptable methodology for conducting a cost-benefit evaluation including the cost/risk averted ratio used.

For the generic BRC dose level it will also be necessary for the NRC to hold a rulemaking hearing to establish the dose level. The rulemaking would focus on the cost-benefit evaluations for all or a significant portion of the regulated practices and activities. In the cost-benefit evaluations performed by the NRC, both the expected and the indeterminate exposure pathways

(as appropriate for recycle or reuse scenarios) would be accounted for in the evaluations. The dose computation models used by the licensees in seeking approval of a proposed practice in compliance with the codified BRC dose level should be essentially identical to the NRC cost-benefit models and would therefore be pre-approved. The basis for applying the generic BRC dose level to a practice or activity which was not included in the rulemaking cost-benefit evaluations is not clear, unless the cost-benefit evaluations were conducted with an excessive degree of conservatism to account for unidentified practices or sources.

SUMMARY

Overall, the above definitions for de minimis, BRC and generic BRC are internally consistent and consistent with the fundamental principles of radiation protection and dose limitation. With the exception of the term BRC, the above definitions are reasonably consistent with the use and applications of the terms by various agencies in various contexts. For the term BRC, the concept that the dose level should be cost-benefit based has been lost in many of the uses of term, which in some degree can be attributed to the unfortunate choice of the term "below regulatory concern." The inclusion of the cost benefit consideration in determining a BRC dose raises a question regarding the validity of establishing a broad-based generic BRC because of the nature of cost-benefit evaluations. However, if the generic BRC dose level is not established too conservatively, it could still be useful for licensees not electing to petition the NRC under the existing BRC policy statement. The ultimate benefit of a generic BRC dose level is that it could, in practice, become a de minimis dose level. It would appear that items such as multiple sources and indeterminate exposure pathways can be accounted for in the implementation portion of the regulation. For the nuclear power industry for waste disposal and unrestricted release of materials, it would appear that the greatest benefit would accrue from rulemakings which (1) established a de minimis dose level, and (2) implemented the BRC policy statement by the NRC's adoption of a source-specific cost-benefit-based BRC limit for all power plant wastes considered together as one composite. A generic BRC dose level is not likely to provide a significant benefit for activities such as waste disposal and unrestricted release of materials. Other nuclear power licensee's activities could have a greater benefit from the establishment of a generic BRC dose level.

Criteria for Release of Decommissioned Nuclear Facilities for Unrestricted Use

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INTRODUCTION

A recent report by the U.S. General Accounting Office cited the need for residual radiation criteria for nuclear facilities decommissioning. Government is emphasizing public health and environmental protection in the operation and remediation of facilities. At the same time, an increasing number of private sector facilities will be ready for decommissioning. Effective and workable criteria on which to base the decision to release nuclear facilities, as well as associated equipment and materials, for unrestricted use are clearly important. This workshop itself lends further testimony to the need for such criteria.

Release criteria must provide suitable protection for public health and safety and the environment. They must also be workable in the sense that they can be implemented and monitored effectively. Today I would like to bring you a perspective on unrestricted use release criteria from the viewpoint of an owner, operator and decommissioner of a nuclear facility who is subject to the regulatory or policy oversight of three federal and two state agencies, as well as county and city government units.

First I should be specific about what I mean by unrestricted use. I am using the term literally -- free from any restrictions on future use. By this I mean that a building is suitable for use as a cafeteria or laboratory without any restrictions; that a piece of furniture is suitable for use in an office or for donation to a day care center without any restrictions; that items of equipment are suitable for donation to a school or for recyclable scrap without any restrictions; that a piece of land is suitable for a playground or for residential development without any restrictions. This absence of restrictions means no further control or monitoring, regulatory or otherwise, over what is done with the released property.

DECOMMISSIONING AT BATTELLE

At Battelle, we are currently dealing with the decommissioning of all or part of 15 buildings that have been used for nuclear materials research at various times over the past 46 years. The facilities are operated under a U.S. Nuclear Regulatory Commission (NRC) license, and are also subject to the relevant U.S. Department of Energy Orders as well as applicable regulations of the U.S. and Ohio Environmental Protection Agencies. Most of the affected buildings are currently used for non-nuclear research, and all are expected to be used for R&D offices and laboratories following decommissioning.

The facilities include a hot cell laboratory which was used until recently for post-irradiation examination of power reactor fuel elements, and a former research reactor which was partially decommissioned in 1978. The remaining 13 buildings were used for R&D on properties and fabrication methods for uranium and thorium, as well as for research with radiotracers.

The residual radioactivity in these buildings is primarily low levels of natural uranium and thorium. The residual radioactivity is contained within the buildings, and no hazardous or radioactive materials were buried or otherwise disposed of at the site. In short, this decommissioning is far less dramatic than some of the decontamination activities being undertaken elsewhere. However, it is probably representative of a large fraction of future decommissioning projects.

RELEASE OF BUILDINGS AND EQUIPMENT

Annex C in fuel cycle facility licenses is essentially the non-reactor equivalent to Regulatory Guide 1.86, and both are similar to proposed ANSI Standard N13.12. Annex C (or Reg Guide 1.86) is extensively used as the basis for releasing buildings -- as well as equipment and material -- for unrestricted use. A close reading of Annex C suggests that it is intended primarily for application to a scenario involving continued non-residential use of the building. It does not appear to be applicable to all plausible future uses of a building.

For example, consider a laboratory building with interior surfaces contaminated with natural uranium. If this building is released for unrestricted use according to the acceptable surface contamination levels of Annex C, then the interior surfaces of the building could average 5000 dpm/100 cm^2. A little arithmetic shows that, to a first approximation, dpm/100 cm^2 is approximately equal to pCi/g for uranium and thorium in a thin surface layer. This implies surface contamination of 5000 pCi/g, which is equivalent to about 0.7 weight percent uranium in the surface layer. Suppose that several years after release of the building a new owner, in converting it for office use, decides to clean up the walls and floors by removing a thin surface layer. In so doing, he creates a particulate waste material containing a few tenths of a percent of uranium by weight. This concentration of uranium is licensable under 10 CFR 40.13. Of course, the new owner knows nothing about uranium waste -- the building was, years before, released for unrestricted use. As a result, a few tons of licensable source material may end up in a local landfill. Suitably treated and packaged, this material is unlikely to pose a threat to public health or the environment. However, if handled or disposed of in a

condition under which the waste material could become airborne, it is likely that air concentrations of uranium in excess of the limits of 10 CFR 20, Appendix B, could occur.

Annex C also does not appear to have been generated for application to unrestricted release of equipment and materials from the decommissioned facility. Here the principal concern is the likelihood that the released item will find its way into use by the general public. For example, such items may be reused by the current owner, sold or donated to other organizations, or disposed of as scrap with subsequent recycling. The lack of control after release together with the high probability of public contact, suggest very conservative criteria should be used for release. The logical alternatives are either no detectible radiation above background (generic criterion), or levels low enough that pathways analysis shows the exposure to the general public to be below the basic dose limit (situation-specific criteria).

Clearly generic criteria are much simpler to implement and monitor. However, they run the risk of being pegged to "worse case" and hence being too restrictive (and costly) for many situations, or of being pegged to some "typical case" and hence not restrictive enough for some situations. The situation-specific approach has appeal because release criteria can be pegged to a risk-based Basic Dose Limit. The important question is "What Basic Dose Limit do I use?" Limits ranging from 4 to 100 mrem/year above background are being proposed by various groups. Adopting a uniform and defensible Basic Dose Limit is crucial to any credible criteria, be they generic or situation-specific.

SOIL CRITERIA

Generic soil release criteria for radium and thorium are available in 40 CFR 192. Situation-specific criteria must be used for other nuclides. Typically this takes the form of a site-specific pathways analysis which assesses the annual dose to a maximally exposed member of the general public under an appropriately conservative scenario. Hydrological and geological parameters are highly variable and highly site specific, and the most appropriate future-use scenario is also dependent on the site location, ownership, and other factors. For these reasons, the situation-specific approach is appropriate for soil release criteria.

As I noted above, a uniform and defensible Basic Dose Limit is essential to credible criteria. For soil release criteria, a generally accepted pathways analysis model is also important. Such models are available and others are under development. From the viewpoint of an owner, operator, and decommissioner of a nuclear facility, the availability of a personal computer based model which is acceptable to all cognizant regulatory bodies is especially important. In combination with a uniform Basic Dose Limit, this would greatly enhance the implementation of site-specific release criteria.

AIR AND WATER RELEASE

Criteria are provided in 10 CFR 20.106 for concentrations of nuclides in air or water released to unrestricted areas. In addition, 10 CFR 20.303 provides criteria for release of water into sanitary sewer systems. The criteria are nuclide-specific and criteria for combinations of nuclides are provided.

CONCLUSION

In summary, appropriate generic criteria for release of buildings and associated equipment and materials for truly unrestricted use are not available. The prudent approach is to use situation-specific analysis, taking into consideration specific nuclides, concentration levels, and plausible future use of facilities, equipment and materials.

Situation-specific pathways analysis is also appropriate for soil release criteria. Implementation of the situation-specific approach would be enhanced by the adoption of a uniform and defensible Basic Dose Limit, and by the availability of a generally accepted pathways analysis model.

Albert Einstein once said "Everything should be as simple as possible, but no simpler." We should keep this in mind as we further develop criteria for the release of decommissioned nuclear facilities for unrestricted use.

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Session V:

Recycling of Materials and Equipment

A Research Program on the Recycling of Decommissioning Materials at JAERI

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ABSTRACT

Research on reuse of wastes from reactor decommissioning has been done at JAERI since 1987. This paper describes a fundamental research program on reuse of radioactive metal wastes, and discusses future plans for the development of reuse techniques as a realistic method of waste disposal. With respect to the first of these, the melting testing program for radioactive metal coming from the decommissioning of the Japan Power Demonstration Reactor (JPDR) is going to reveal the material balance and the radioactivity balance during melting. The second is at the stage of designing research programs intended to demonstrate real reuse of decommissioning materials.

INTRODUCTION

The generation of electricity by nuclear power in Japan has increased remarkably in recent years, with 36 commercial nuclear power plants (about 28,000 MW) in operation as of the end of November, 1988. But, the life of a nuclear power reactor is only about 30 to 40 years. It is predicted that the decommissioning of commercial nuclear power plants in our country will begin in the second half of the 1990's. The report "Long-term Program on Nuclear Energy Development and Utilization," issued by the Atomic Energy Commission in June, 1982 provided concrete guidelines for decommissioning nuclear power plants. Because of our country's small size and insufficient space to construct new nuclear power plants, the report recommended, as the basic decommissioning policy, quick dismantling (5 to 10 years) after shutdown, for the reuse of the site for follow-on reactors. The decommissioning program of the Japan Power Demonstration Reactor (JPDR), under progress at the Japan Atomic Energy Research Institute (JAERI), is based on this basic policy, and aims both at the development of reactor dismantling technologies and at the improvement of dismantling safety.

It is expected that decommissioning a nuclear power plant will generate a large volume of dismantling waste. For example, it is estimated that dismantling a 1,100 MW nuclear power plant will result in 500 to 550 thousand tons of waste (including 40 to 50 thousand tons of metal) within a concentrated period of time. To carry out the decommissioning smoothly, the establishment of a rational system for treating and disposing of the waste is essential. If we can use the dismantling waste as a resource, this can decrease the amount of waste in a very

meaningful way. In addition, this direction is in harmony with national policy; the reuse of extremely low-level radioactive solid wastes is suggested in the report "Philosophy of Safety Regulation for Land Disposal of Low-Level Radioactive Solid Wastes" submitted by the Nuclear Safety Commission in October, 1985. But dismantling wastes from nuclear installations differ from the usual industrial wastes, and safe and rational techniques for recycling them have not been established yet. From this viewpoint, JAERI started its reuse research with the JPDR decommissioning materials in parallel with investigations of the technical problems for reusing the dismantling waste safely as resources. The objectives of this reuse research are to establish the techniques for reusing the dismantling waste from decommissioning nuclear power plants, and to contribute to the discussion on technical guidelines for reuse.

PRELIMINARY SURVEY ON DISMANTLING WASTE REUSE

In order to establish a system for safe and economical reuse of dismantling wastes, development of the uses according to characteristics (material, form, radioactivity level, etc.) of the waste is indispensable. The results of a preliminary survey on the uses of metal and concrete waste are demonstrated in the following sections.

Reuse of metal waste

The system (treatment technique, distribution route of treatment and recovery, etc.) for the reuse of usual industrial metal waste is already established, because metal waste has inherent economical value. Therefore, it is expected that the existing reuse system will function well for the waste under an exemption level. The exemption level has not yet been determined in Japan. Although the exemption level to be decided upon in the future may exert influence on radioactive metal waste reuse, the development of the possible uses of radioactive metal will play an important role. From this viewpoint, with the ease in promoting reuse taken into account, the uses of metal waste for applications in controlled areas or non-controlled areas within the nuclear installations have been discussed, and for use off-site, on the condition that remanufactured articles are confined to a certain region. The results to date are summarized as follows:

Promising uses in controlled areas
- Reinforcing bars and other structural materials for building
- Tanks
- Casks, canisters and waste containers, etc.

Promising uses in non-controlled areas
- Reinforcing bars
- Pipings, etc.

Promising uses off-site
- Base materials (caissons, piles, etc.)
- Reinforcing bars
- Pipelines, etc.

Reuse of concrete waste

As base information for discussing the uses of concrete waste coming from the dismantling of nuclear installations, the present situation of concrete waste coming from the demolition of usual structures was investigated. A survey was conducted on the annual consumption and disposal of concrete in our country, the treatment and disposal processes of concrete, and their costs. Discussions on the uses for concrete and the solutions to remaining problems were carried out on the prospects of reusing concrete wastes. According to the survey, the amount of concrete waste disposed of in 1986 is equivalent to the concrete wastes from dismantling 30 to 40 1,100 MW nuclear power plants. Seventy percent of the concrete waste was disposed of, and only the remaining thirty percent was reused. Even if concrete waste is reused, its applications are restricted to roadbed or backfilling materials. However, it is anticipated that in the near future, disposal means like backfilling will be difficult to use, owing to land shortage and protection of the environment.

Although most of the uses are not characterized well, some typical examples are as follows:

- Structures of nuclear installations (structural wall, shielding wall, sectioning wall)
- Waste disposal pit and container
- Roadbed material
- Cementation material

It is essential that the safety, economics, and quality of these applications are thoroughly examined, and that a rational reuse system is established through the development of the necessary technology.

MELTING TESTING PROGRAM FOR RADIOACTIVE METAL

There are two ways to reuse metal dismantling wastes. One is to reuse contaminated equipment (pump, pipings, etc.) as it is after decontamination in other nuclear installations. The other is to produce remanufactured articles (waste container, construction materials, etc.) by melting the metal, and utilizing them within nuclear installations or in another place. With respect to the latter case, the behavior of radionuclides, during melting and solidification of the radioactive metals, must be revealed for the safety of workers and the public. But, at present in Japan, there is no relevant data. Therefore, JAERI has started research titled "Reuse Technology Development of Low-Level Radioactive Wastes", performed under contract with the Science and Technology Agency since 1987. In this research, tests of the melting of radioactive metal waste from the dismantling of the JPDR, and of other materials, are planned. The objectives of these melting tests are to investigate and assess the movement of radionuclides during solidifying processes and melting, and its influence on the environment. The schedule of the program is shown in Fig. 1.

Experimental equipment

The production of the experimental equipment to be used in the melting tests of radioactive metals will be completed by 1990. The schematics of the melting equipment and the flow of work procedures are shown in Fig. 2.

The melting equipment consists of a melting furnace, casting equipment, sampling apparatus, filtering equipment, radiation monitors, etc. A high frequency induction furnace (500 kg capacity) was considered most appropriate for the kind of work to be undertaken. Some of its positive characteristics are:

a) Adequate homogeneity of radioactivity in melted metal
b) Feasible melting of carbon steel and stainless steel
c) Little secondary wastes
d) High reliability, etc.

The melting furnace will be surrounded by a containment chamber to minimize the spread of contamination by radionuclides during operation. The filtering system to collect dust is composed of a cyclone, bag filters and a High Efficiency Particulate Air (HEPA) filter.

Contents of testing

Fundamental melting tests with radioactive metals will be performed as a function of materials, condition of radioactivity (activated, contaminated), etc. Accordingly, three types of tests are planned: cold tests, tracer tests with radioactive isotopes (RI), and actual waste tests with JPDR decommissioning materials. Data on the materials balance, radioactivity balance, and dose rate in the working environment during the melting and solidifying processes will be collected, as well as the operation characteristics of the experimental equipment. The tests are summarized below.

(1) Cold test

A cold test serves as a trial run of the melting equipment and provides an understanding of the materials balance. Operational data on melting equipment (relation between melting time and temperature, off-gas temperature, etc.) and on the movement of materials (adhesion to the equipment, production of slag and dust, etc.) will be collected.

(2) Tracer test with RI

The evidence suggests that some metal wastes are contaminated with radionuclides which are produced both by the activation of the base metal and impurities in structural materials and by fission products arising from the breakage of fuel rods. The aim of this test is to investigate the behavior of these radionuclides in the melting and solidifying processes.

This is difficult to examine using JPDR decommissioning materials because of its decreased radioactivity. For this reason, the amount of radionuclide movement from melting metal to solidified metal, furnace wall, slag, pipings and off-gas will be measured in commercially available metals coated with radioisotope tracers that imitate the nuclides found in contaminated material of the light water reactor. The radionuclides to be used are Mn-54, Co-60, Zn-65, Sr-85, Cs-137, etc. In addition, the radiation dose rates on the equipment surfaces and in the working environment will be measured, to obtain radiation control data.

Testing parameters and assessment items are as follows.

1) Testing parameter
   a) Materials (carbon steel, stainless steel)
   b) Radioactivity
   c) Condition of radioactivity (activated, contaminated)
   d) Nature of flux
   e) Melting temperature

2) Assessment items
   a) Material movement from charged material to ingot, slag, off-gas, etc.
   b) Movement of radionuclides to ingot, slag, off-gas, etc .
   c) Radioactivity distribution within an ingot
   e) Radiation dose rate in working places
   f) In-air radioactivity of working places

(3) Actual waste test with JPDR decommissioning materials

Contaminated and activated metals such as pipings, valves, and a piece of the pressure vessel from decommissioning the JPDR will be melted. By these melting tests, the safety of melting real radioactive metal will be demonstrated.

FUTURE PLAN

The key to success in decommissioning a reactor is to treat and dispose of large volumes of dismantling waste. In the future, establishing better ways of treating and disposing of waste will become increasingly important. In particular, the economics of treatment and disposal of waste, and environmental protection from waste, will be regarded as major problems. Therefore, we must find a way to solve these problems, from a long-term standpoint, irrespective of the types of waste. The reuse plans for the future are described below and illustrated in Fig. 3.

Production and application of remanufactured articles

Before reusing dismantling waste, whether metal or concrete, it is very important to examine the safety of both the manufactured items and the manufacturing processes. The necessary data, however, does not exist. Therefore, integrated research is planned for both metal and concrete wastes. In the research, feasibility for restricted or unrestricted use of remanufactured articles will be investigated through the production of such articles. For metal waste, a large scale melting facility will be constructed, taking into account the results obtained from the fundamental melting test described earlier. All the processes (melting to forming) will be tested to produce high value-added remanufactured articles (waste containers, etc.) using the metal wastes of the JPDR. As to concrete wastes, remanufactured articles like construction materials made from radioactive and non-radioactive concrete waste or composites with metal will be produced. Also, confirmation testing, intended to examine the performance of these remanufactured articles, will be carried out. Furthermore, buildings such as waste storage facilities will be constructed using some of waste to confirm its actual integrity.

Effective utilization of radioactive waste

Activated metal waste generally must be stored, owing to its high radiation dose rate. This makes treatment and disposal difficult. However, if the radiation energy emitted from the waste is used effectively, the radiation can become a benefit. For example, radiation energy is used widely for chemical reactions, disinfection, food preservation, etc. In addition, utilization of radiation is being tried as a means to purify the environment, such as purification of exhaust gases, decomposition and removal of contamination in sewage water, and disinfection and composing of sewage sludge. Activated metal waste could be utilized effectively in the fields of:

- Disinfection of sewage water, medical apparatus and agricultural products
- Purification of exhaust gases, etc.

Accordingly, the feasibility of reusing highly activated metal waste as a radiation source will be investigated, including the related forming techniques and processes, equipment, and economy.

CONCLUDING REMARKS

JAERI has started research on reuse of decommissioning material. In this paper, the fundamental reuse research program of metal waste is described, together with the future plans including production and application of remanufactured articles. Reuse technology may be an alternative means of disposing waste. The radioactive metal melting test program in progress is expected to bring a great deal of useful fundamental data on recycling metal waste. Similarly, plans described here utilize the vast volumes of dismantling wastes from a decommissioned reactor.

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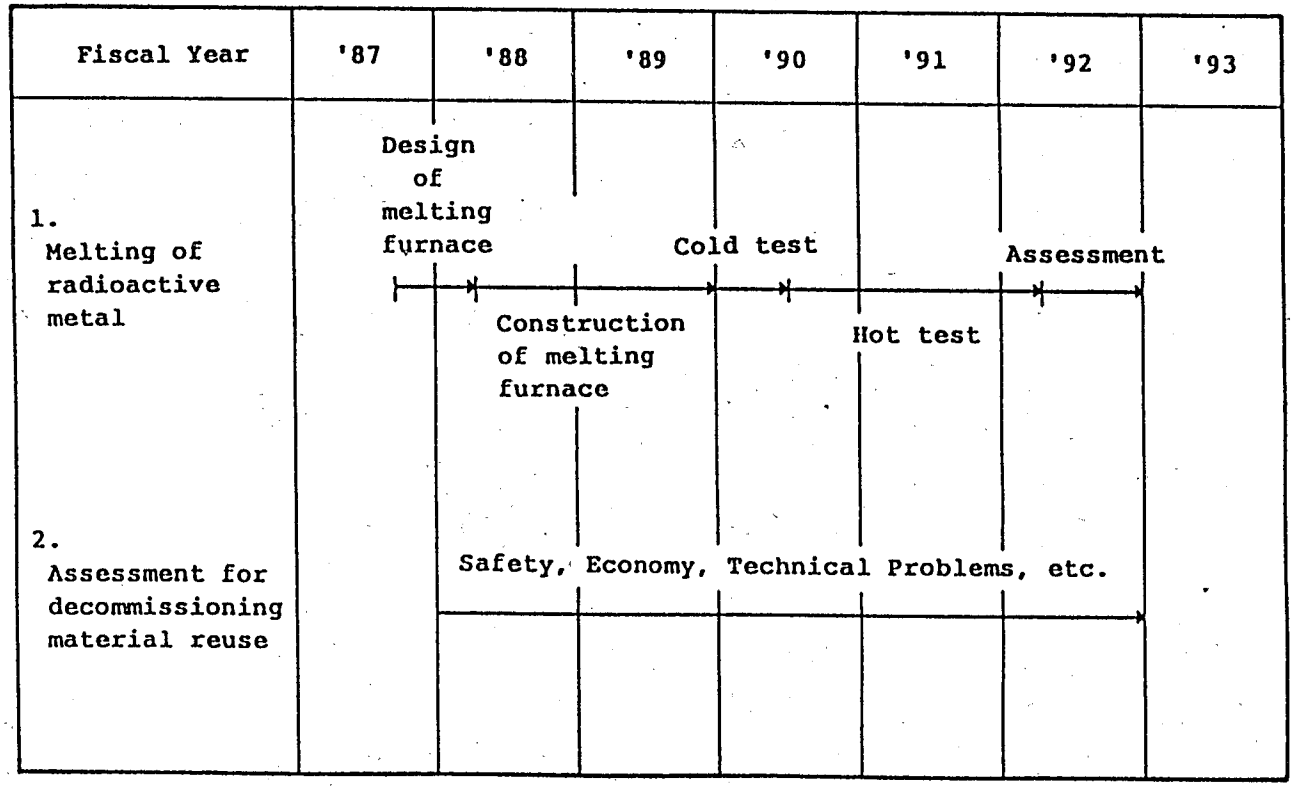

Fig.1 Schedule of radioactive metal melting testing

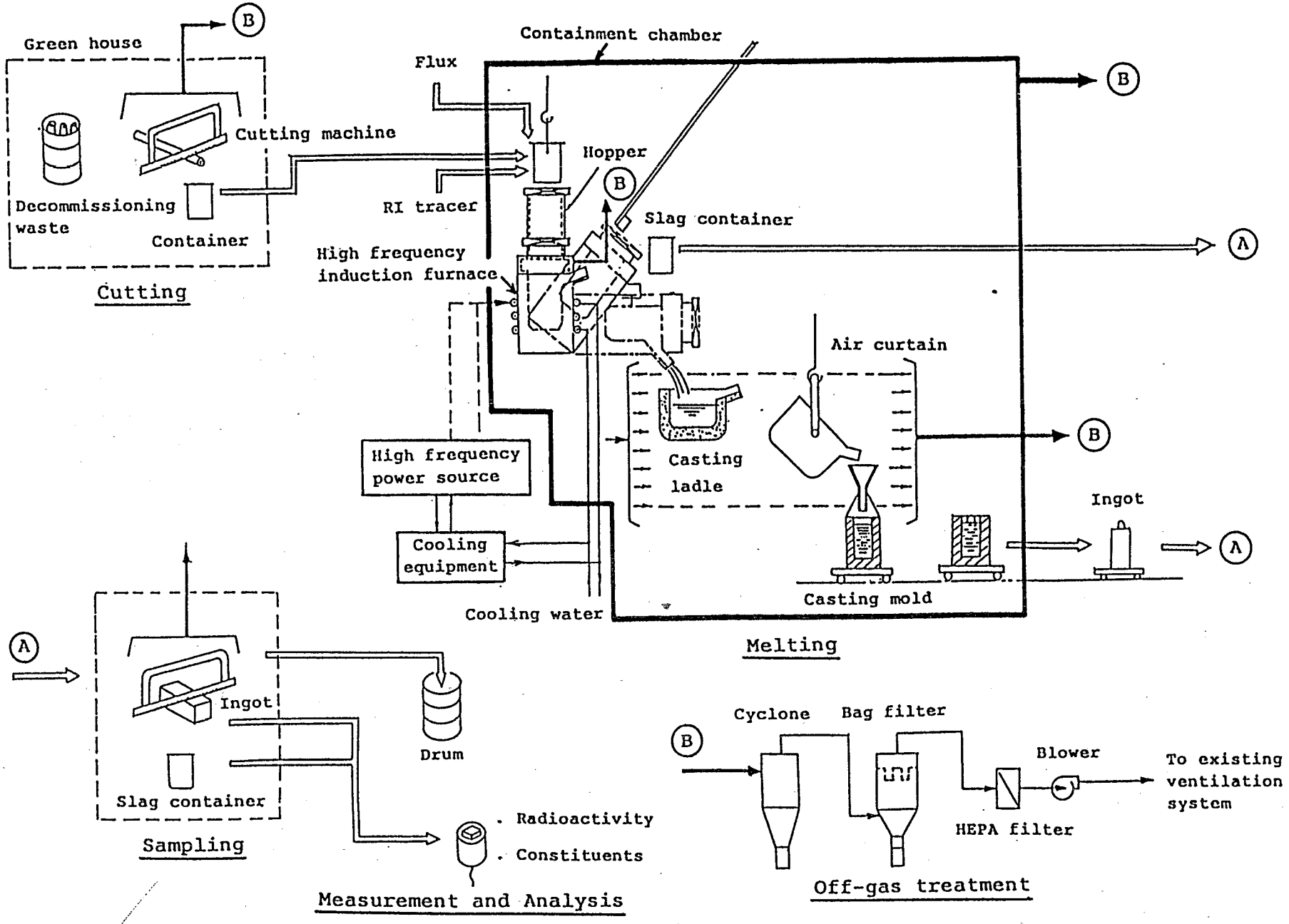

Fig.2 Schematic diagram of melting equipment and flow of work

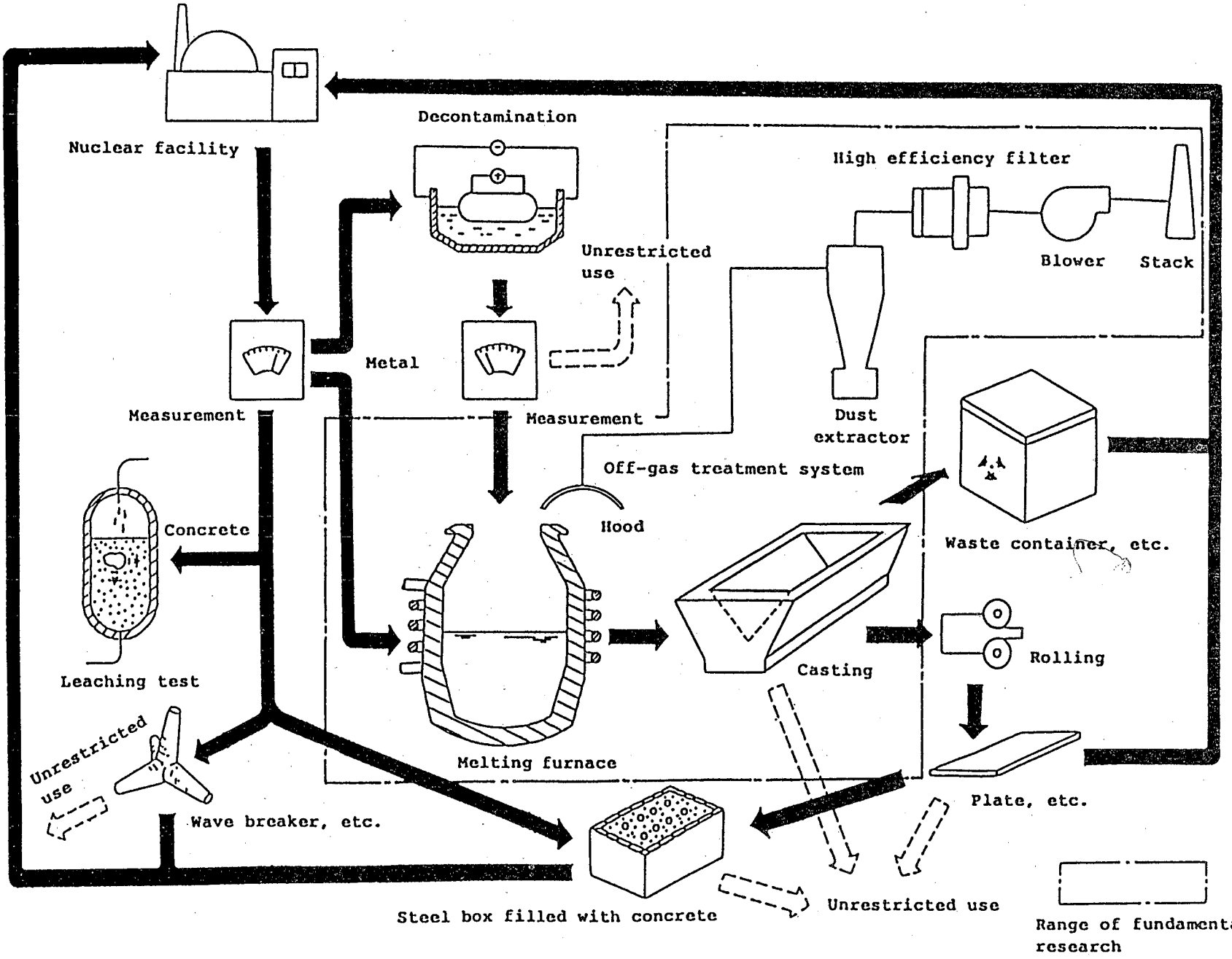

Fig.3 Flow diagram of decommissioning material reuse

Effects of Residual Radioactivity in Recycled Materials on Scientific and Industrial Equipments

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ABSTRACT

Health effects, as well as scientific and industrial effects, should be assessed in establishing residual radioactivity criteria for recycled materials. In the present study, a comprehensive literature survey has been carried out on the effects of residual radioactivity on scientific and industrial equipment. To develop a method for assessing the scientific and industrial impacts of recycling materials, an investigation also was conducted on measures to reduce these effects, on the trend in rates of production of the equipment, and on the progress of the technologies.

The kinds of equipment most likely to be affected by residual radioactivity are large scale integrated circuits (LSI), photographic films, and low-background radiation counters. Procedures for assessment of the effects of residual radioactivity on these have been suggested. Recycling of contaminated materials is expected to bring both gains and losses. An optimum residual radioactivity level was proposed as an index for the assessment of the effects. The optimum residual radioactivity level is defined as the level at which the difference between economic benefits and costs from recycling of contaminated materials is maximum. We discussed the relationship between the optimum residual radioactivity level and a residual radioactivity level that will be derived to produce no significant radiological hazard.

INTRODUCTION

The decommissioning or remodeling of nuclear reactor, nuclear fuel facilities, particle accelerators and radioisotope handling facilities generates large amounts of slightly radioactive material and equipment. The establishment of residual radioactivity criteria (RRC) which enable us to reuse or recycle such material and equipment is absolutely essential for the exploitation of extremely low-level radioactive wastes as potential resources, as well as for the reduction of their volumes.

The RRC will ensure that the health effects will be negligibly small from recycling contaminated materials. When is it sufficient to consider only health effects?

Modern science and technology have been and will be making remarkable progress to change our society. There are various types of high technology industry that can use only extremely high quality materials. But the probability of contaminated materials effecting such technologies is expected to increase.

For example, the trend in large scale integrated circuits (LSI) is towards higher levels of integration. However, a reduction in memory cell size, with a corresponding reduction in signal charge, leads to increases in the sensitivity to noise. May and Woods [1] recognized that alpha particles from radioactive impurities in the packages could cause a soft error in dynamic RAM and CCDs. Therefore, the LSI industry has been seeking extremely high quality material as a constituent material.

This phenomena indicates that scientific and industrial effects as well as health effects should be considered in developing RRC. Assessments on health effect from residual radioactivity in recycling material have been conducted by the Commission on European Communities (CEC) [2] and the U.S. Nuclear Regulatory Commission (NRC) [3]. However, an assessment of scientific and industrial effects has not been carried out.

In assessing health effects from radiation quantitatively, there are many procedures to be investigated; analysis of exposure pathways, identification of critical pathway and development of assessment models. These same procedures should also be investigated when assessing the industrial effects from residual radioactivity.

This is a pilot study on the effects of residual radioactivity on scientific and industrial equipment. Its purpose is to obtain background information to establish the residual radioactivity criteria. The study was started from 1988 as a three-year program.

1988: Investigation of the effects of residual radioactivity on scientific and industrial equipment.

1989: Development of a calculation code for assessment of the scientific and industrial effects of recycling contaminated materials.

1990: Assessment of the effect on possible scientific and industrial products by using the calculation code.

In the present paper we describe the effects of residual radioactivity on scientific and industrial equipment and the development of an assessment method.

THE EFFECTS ON SCIENTIFIC AND INDUSTRIAL INSTRUMENTS

With unrestricted recycling, contamination levels in the material may be very low, so it is important to consider the effects from slight contamination.

The rapid progress of science and technology will increase the possibility of residual radioactivity affecting the scientific and industrial fields, in particular, new technologies such as LSI, and high sensitivity photographic film.

The investigation of the effects was conducted to cover all kinds of instruments. The investigation was focused on the following items: phenomena and mechanism of the effect, occurrence rate of the effect, occurrence rate from other causes, measures to reduce the effects, and the trend of production.

Large Scale Integrated Circuit (LSI)

(1) Effects of radiation on LSI

Radiation effects on LSI consist of the single event, and the total dose effect. Furthermore, the single event consists of soft error and lutchup.

Total dose effect:

Ionizing radiation produces stored charge and oxide interface states on the chip surface. MOS technologies can seriously be affected, since they involve surface effect devices. The major effects are a shift in threshold voltage of the transistor, and an inversion of p-type regions under field oxides. This effect is a semipermanent phenomena and can be evaluated in terms of the cumulative exposure which causes the failure of LSI function. The dose level depends on the type of device and integration density. The cumulative dose which leads to the effect is from 10^3 to 10^4 rad for nMOS-DRAM, nMOS-SRAM, CMOS-SRAM [4, 5, 6].

Lutchup:

Lutchup is a parasitic phenomenon occurring in CMOS structures due to feedback in the p-n-p-n structure. Lutchup occurs when currents caused by radiation in the substrate, or well, forward bias a diffusion-background junction. Once turned on, the current flow will continue until power is removed or until the device is damaged. Lutchup is induced by irradiation of high energy cosmic rays. The error rate of lutchup is one hundredth to one thousandth of soft error rate [6, 9].

Soft error:

Soft error is the upset of stored data by the passage of alpha particles through the memory array area. There are two modes of upset in LSI, memory cell mode and bit-line mode, as shown in Figure 1. In the cell mode, an upset is caused by noise electrons flowing into a storage capacitor. In the bit-line mode, an upset is caused by noise electrons flowing into the n+ diffused layer in the floating bit-line in a read cycle.

The relation between soft error rate (SER) and alpha flux indicates that the SER is proportional to alpha flux, as shown in Figure 2 [1,8]. This figure also indicates that the soft error rate increases as the integration density of LSI increases. Figure 3 shows that devices with smaller critical charges exhibit higher SER [1]. It is known that soft error is scarcely induced by beta-rays or gamma-rays [1].

From the above review, it becomes clear that the soft error is the most important effect induced by radiation emitted from contaminated recycling material.

(2) Factors affecting soft error and reduction techniques

SER depends on critical charge, cycle time, power supply voltage, integration density of LSI and flux, energy, injection point and angle of alpha particle [7, 11, 12, 13].

The information densities of new generations of DRAM have increased four-fold every 3 years since the introduction of a 1k DRAM in 1974 [14]. 64 M DRAM devices are expected to be produced by the year 2000. As mentioned above, the higher integration density tends to have small critical charge, which easily causes soft errors in a DRAM. These trends indicate that soft errors will be very important problem in future.

Many techniques to reduce the soft error rate have been developed. Information on the reduction techniques is essential to estimate soft error rate and the cost of the reduction. Many measures such as reduction of alpha-ray producing impurity in the material, overcoat, error detecting, error correcting codes, and improvement in the design of the circuit or structure of LSI are being developed [7, 15].

(3) Materials used in LSI

Figure 4 shows the cross sections of three types of IC packages: the stacked ceramic type package, the pressed hermetic ceramic type, and the plastic molding type [16]. The LSI parts likely to be contaminated with recycling materials are the ceramic (Al_2O_3), cap (Fe-Ni), steel (Au-Sn) and lead (Au-gild) for the static type, and ceramic and lead for the pressed hermetic type, and lead for the plastics type.

The major metallic materials used in LSI are Al, Au and Cu.

(4) Trend of production

Information on production and application of LSI in society is important to assess the economic impact of recycling contaminated material. Figure 5 shows a trend in the yield of DRAM in terms of cumulative bit number of devices [16]. The yield of 64k DRAM, reached a maximum in 1984. Thereafter, the production of 265k DRAMs increased. Today, the 1M DRAM is a major device. The number of each device changes year by year, but the logarithm of the total number of bits of a DRAM increases in an approximately linear fashion, by a factor of 1,000 times every ten years. The logarithm of the number of DRAMs also increases as a linear function of the year, at a rate of 10 times every 10 years.

Figure 6 shows industrial fields using ICs, and the cost fraction of an IC [13]. In Japan, an application fraction of ICs for industrial products is large. The demand for ICs from office automation systems, like personal computers, is expected to increase in the future. The fraction of IC cost to total cost of the instrument for all instruments is expected to increase. For example, the fraction for an automobile will increase from 2.4% in 1985 to 8.1% in 1990.

The ORRL is an index in establishing the RRC. For example, no net benefits are obtained from Case 3. In Case 1, recycling material contaminated with radioactivity at the level derived from health effects may be advisable. In Case 2, the ORRL may be desirable for recycling.

The ORRL can be estimated using system engineering methods such as non-linear planning. A calculation code is under development to estimate the ORRL and to assess quantitatively the profits and costs from recycling of contaminated materials generated from decommissioning.

CONCLUSIONS

In establishing RRC, not only the health effects but also the industrial effects from residual radioactivity should be considered. Many assessment methods of health effects have been already developed by the CEC, NRC, and others. However, the assessment method for industrial effects from residual radioactivity has not been established. In the present study, a literature survey on the industrial and scientific effects from residual radioactivity has been carried out to identify the effects and to develop an assessment method.

The potential fields in which scientific and industrial effects occur from residual radioactivity in recycling material are LSI, photographic films, and low-background radiation counters. Furthermore, countermeasures to reduce the effects, trends of production rate, and technological advances were surveyed.

Based on this information, an optimum residual radioactivity level (ORRL) is proposed as an index for the assessment. An assessment code to estimate the ORRL and to evaluate the scientific and industrial effects quantitatively is under development. Using this code, the industrial effects from residual radioactivity can be assessed in establishing RRC for both unrestricted and restricted recycling.

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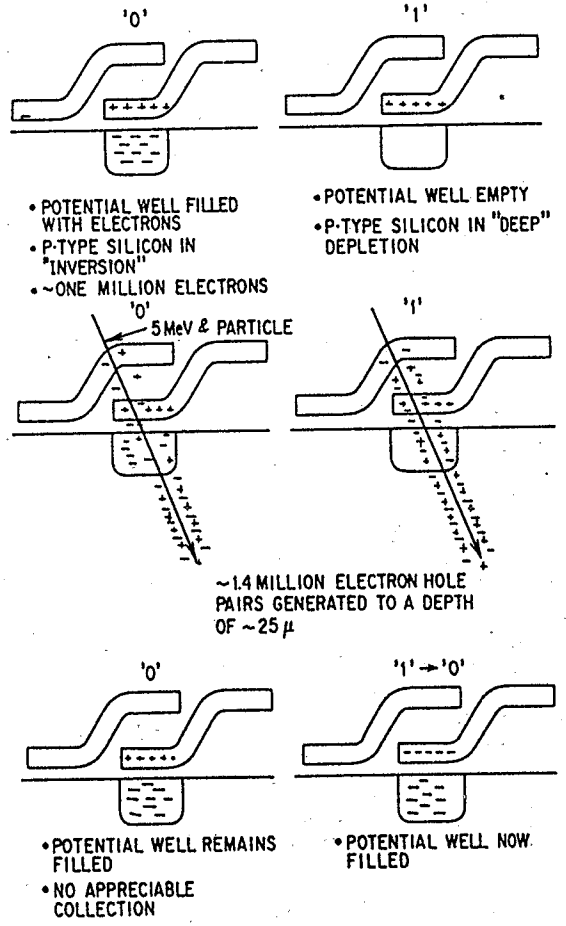
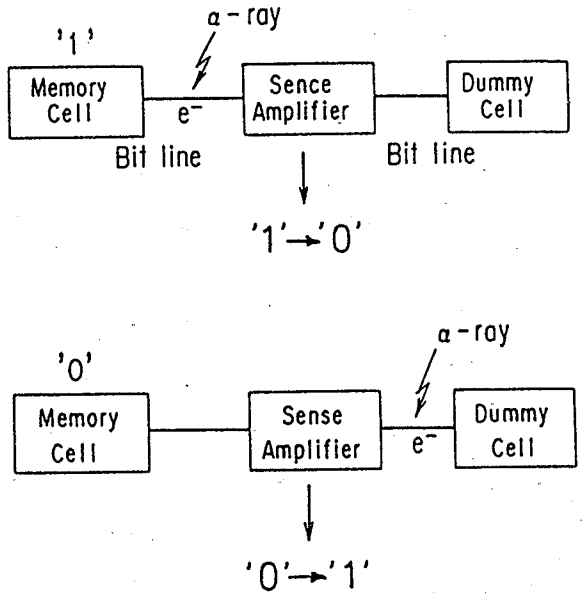


Soft error due to cell contribution
Source : T.C. May and M.H. Wrod (1978)



Soft error due to bit line contribution

Figure 1 Mechanism of soft error in dynamic memory.

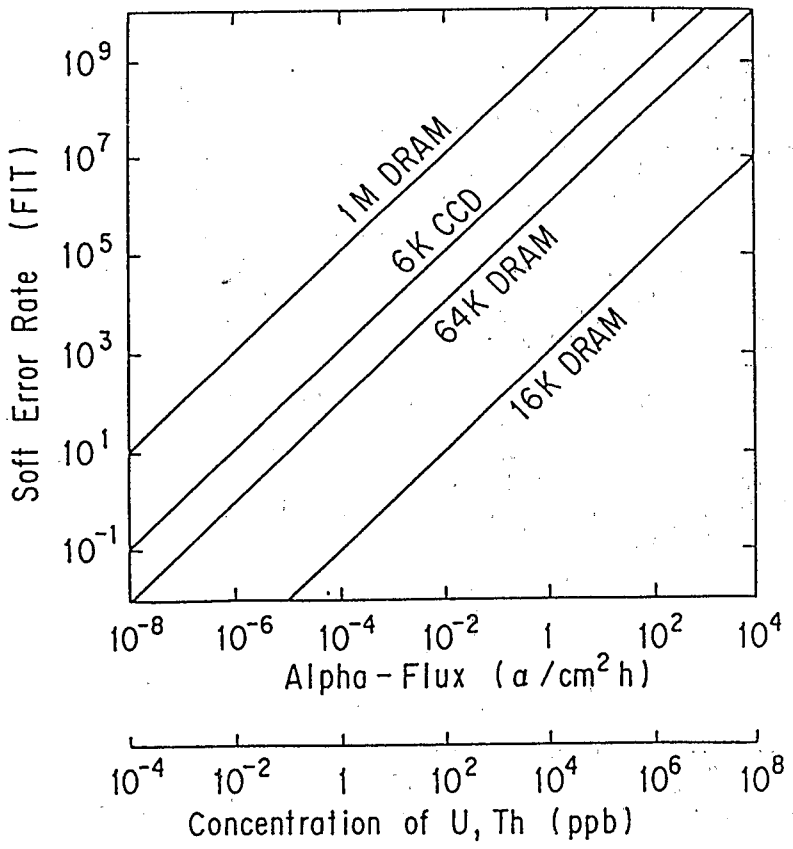


Figure 2 Error rate versus alpha flux, and concentration of U and Th.(Source: Hirai, 1988)

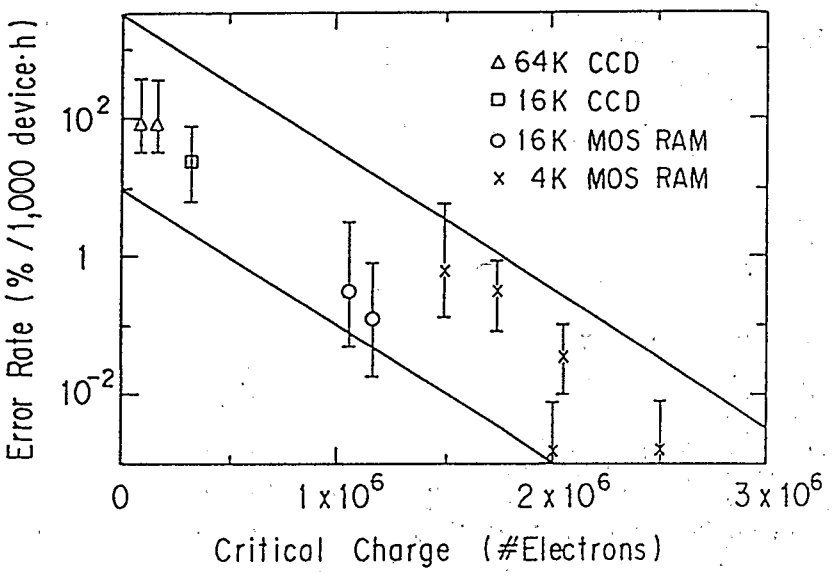


Figure 3 Error rate versus critical change for devices from several manufactures. (Source: T.C.May and M.H. Woods, 1978)

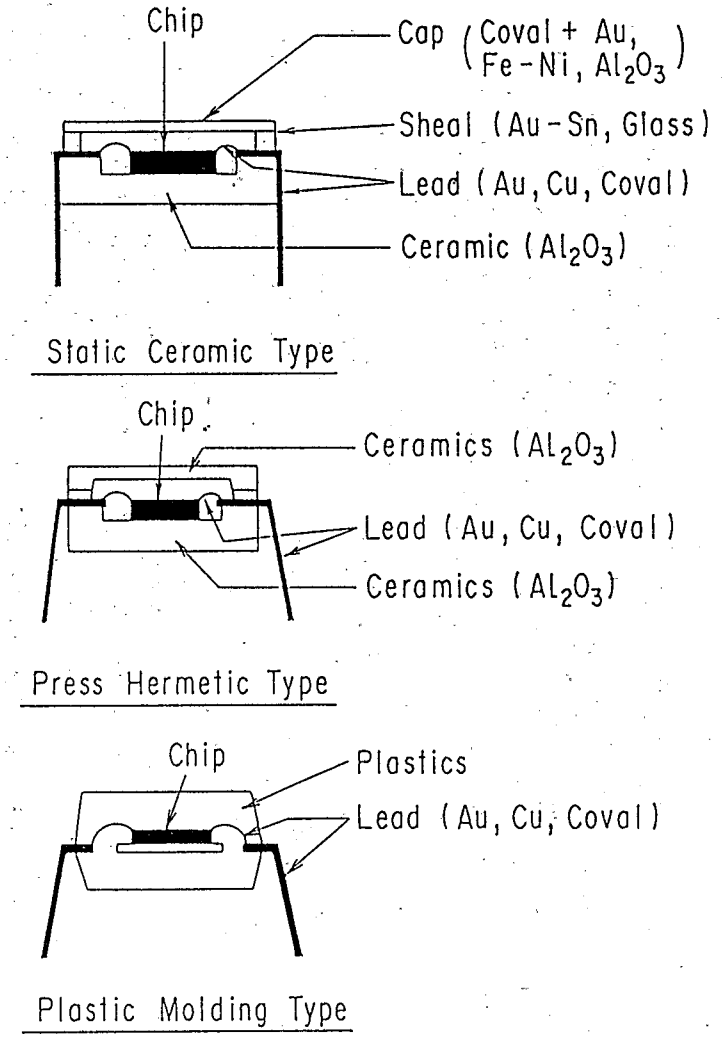


Figure 4 Package types and the constituent materials. (Source: K. Kudo et al. 1982)

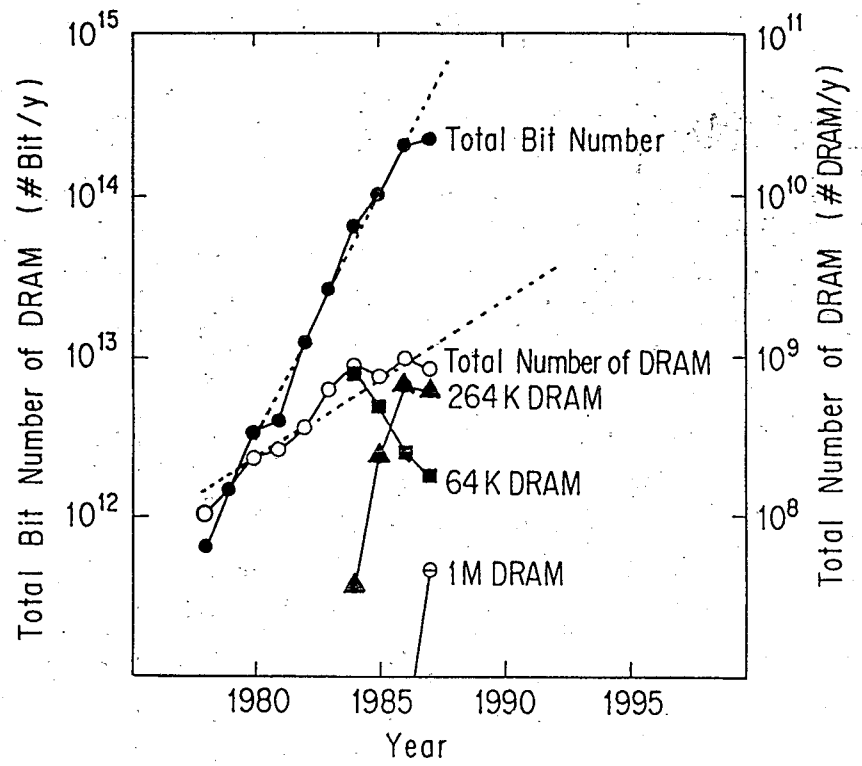
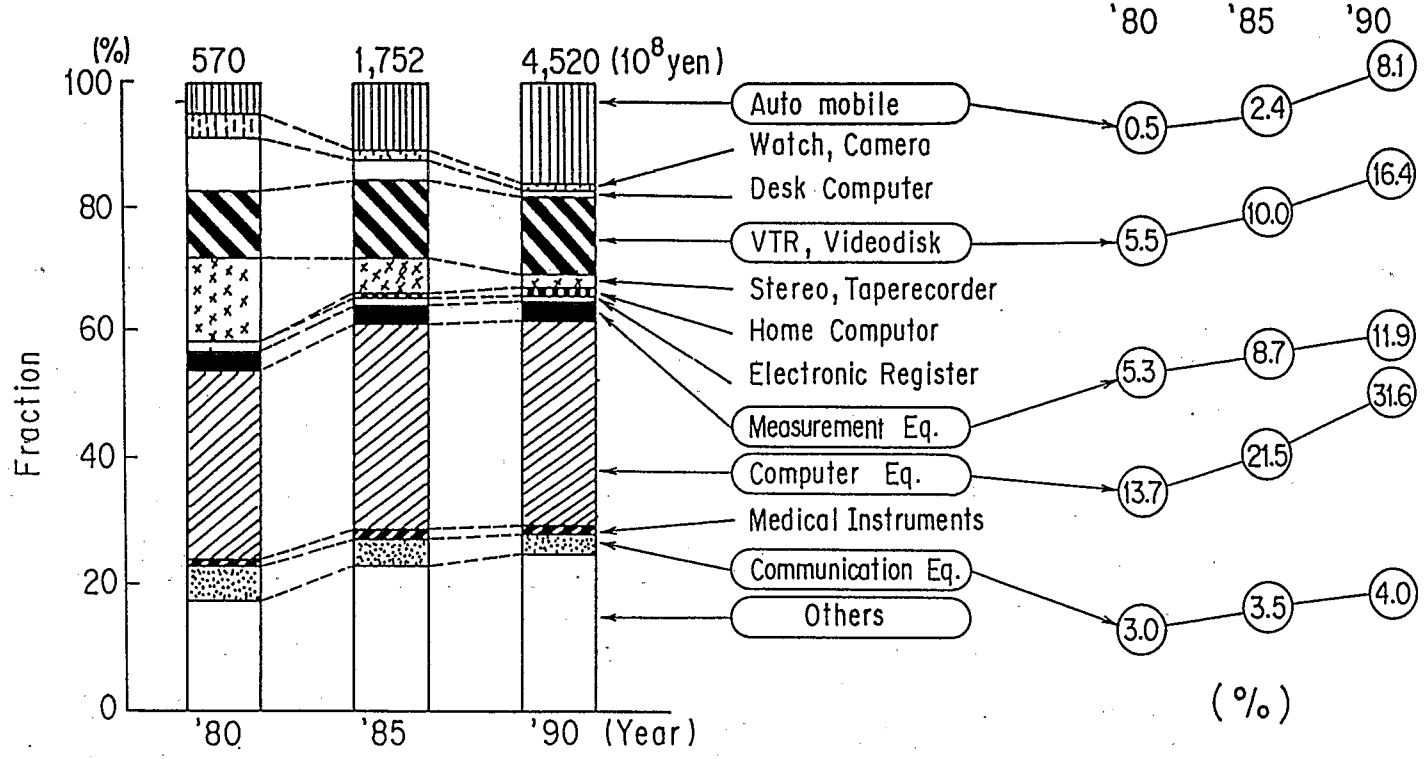


Figure 5 Trend of the production of DRAM in the world. (Source: NIKKEI ELECTRONICS, 1987)



Trend of Application of IC to Industrial Goods

Trend of Ratio of the Cost of IC to Total Cost of the Good

Figure 6 Application of IC to industrial goods. (Source: KAGAKUKOGYO No.51, 1987)

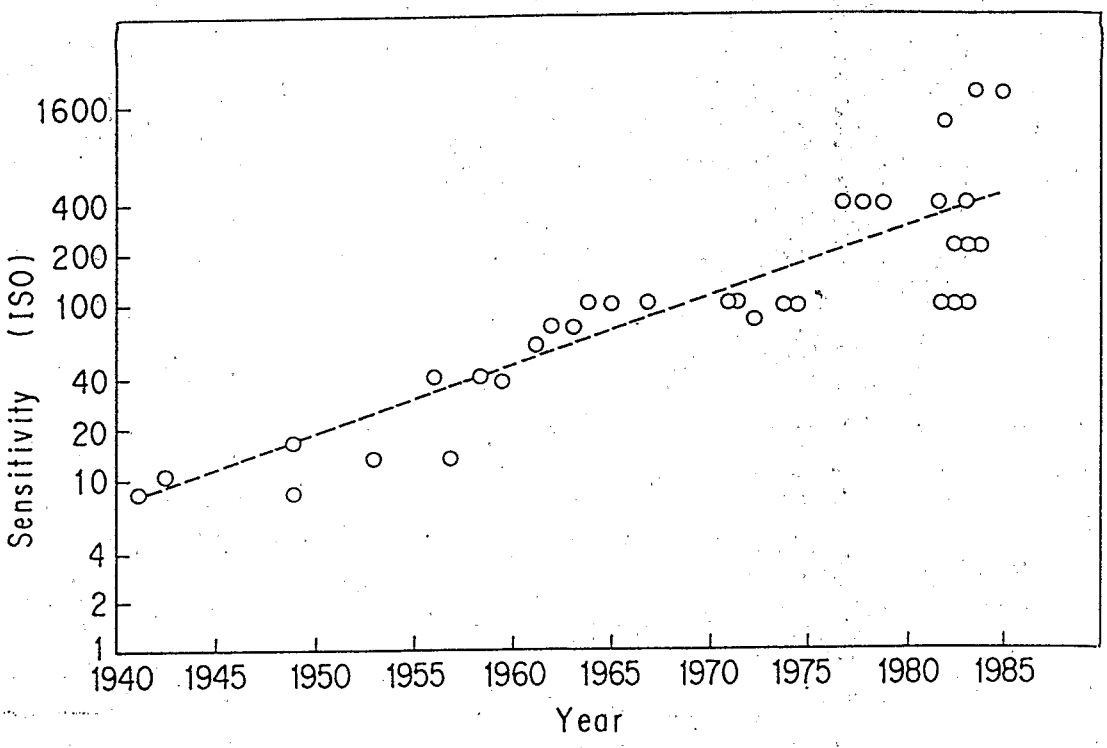


Figure 7 Trend of sensitivity of photographic negative color film. (Source: S.Takada, 1986)

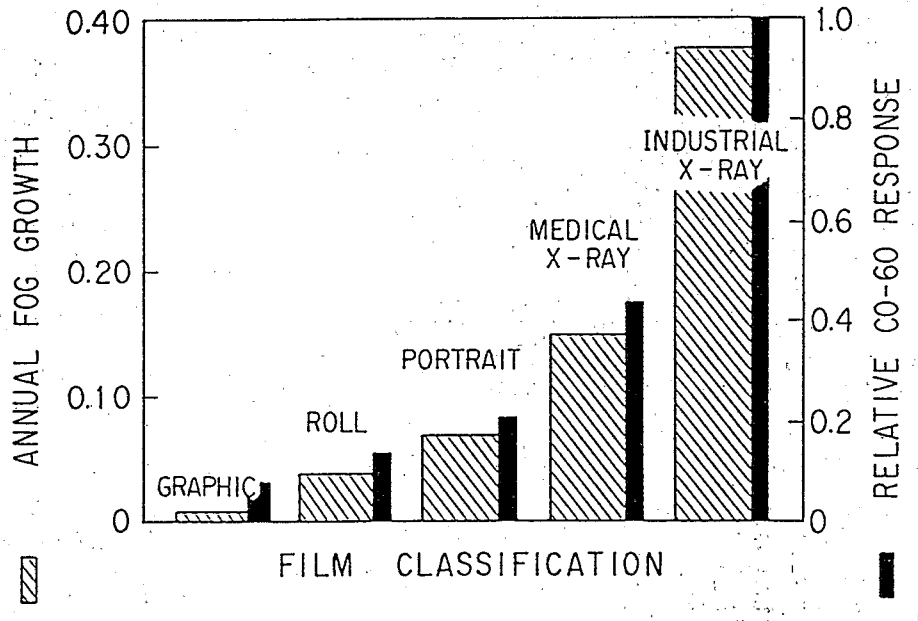


Figure 8 Comparison of normal storage fog glowth per year of five essentially different photographic films, with their corresponding response to Co-60 gamma radiation. (Source: D.P.Jones et al, 1964)

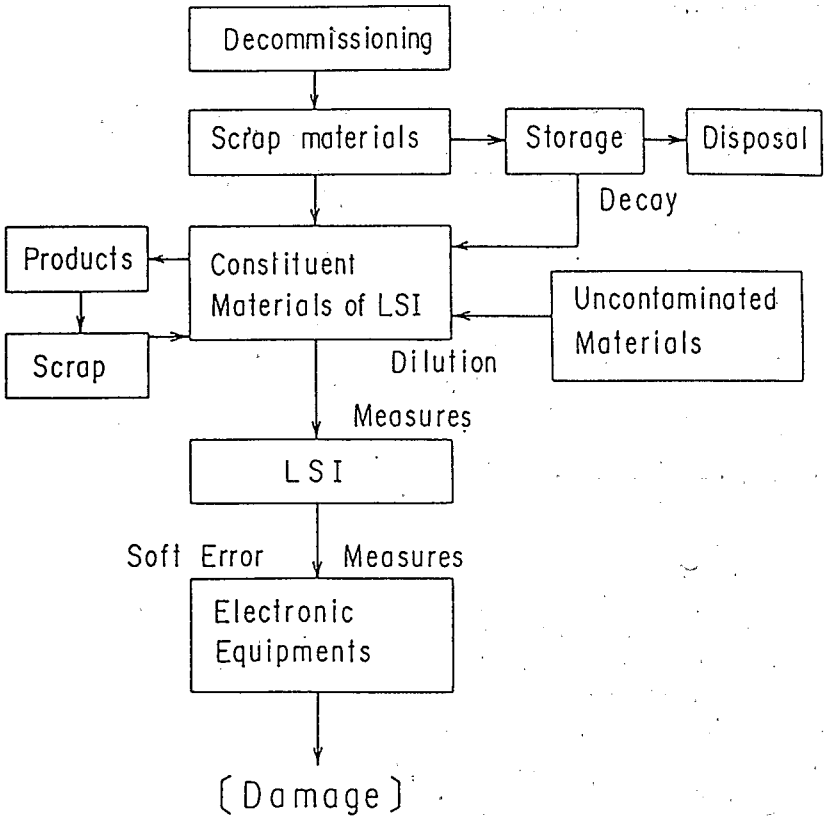


Figure 9 Flow diagram of the effect of recycling materials on LSI

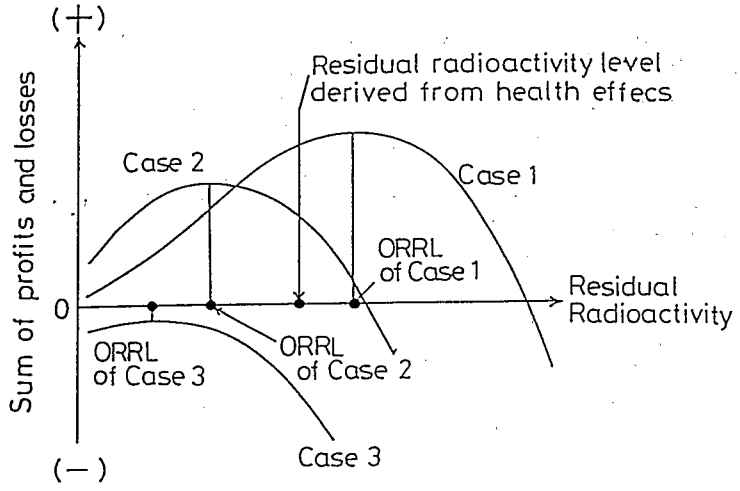


Figure 10 Optimum residual radioactivity level(ORRL) in recycling of contaminated materials and the relationship between ORRL and the residual radioactivity level derived from health effects.

*Development of International Exemption Principles for Recycle and Reuse**

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ABSTRACT

For the past several years, the International Atomic Energy Agency (IAEA) has been investigating the possibility of exempting certain radiation sources and practices from regulatory control. Initial efforts were conducted under the general heading of de minimis. Interest in this topic arises from international recognition that a significant fraction of the wastes from industry, research, medicine, and the nuclear fuel cycle are contaminated to such low levels that applying regulatory controls may be unwarranted. The IAEA evaluation has been conducted by Advisory Groups considering two interrelated topics: to establish principles for exemption, and to apply the principles to various areas of waste management. The IAEA Advisory Groups developed the criteria from modeled assessments of the potential radiation exposure pathways and scenarios for individuals and population groups following the unrestricted release of radioactive materials. Although the scenarios and models used by the IAEA are necessarily generic, consultants to the Advisory Groups attempted to identify the most important radiation exposure pathways based on available literature. This approach is intended to provide the basic framework for the numerical derivation of generic exempt quantities that would be conservative in most situations. In their evaluations to date, the IAEA Advisory Groups considered: 1) disposal of exempt wastes in a sanitary landfill, 2) disposal of exempt wastes by incineration, 3) recycle of contaminated steel, aluminum, or concrete, and 4) reuse of concrete buildings, tools, or equipment. This paper discusses the current status of the IAEA's efforts on the subject of exemption and presents the generic results expressed as overall exemption limits for municipal wastes and materials for recycle and reuse.

The International Atomic Energy Agency (IAEA) has been investigating the possibility of exempting certain radiation sources and practices from regulatory control as an extension

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of its earlier work in the area of de minimis. Because of the potential value of scrap materials recovered during decommissioning of commercial reactors, and because of national and international efforts to minimize radioactive wastes, exemption criteria for recycle and reuse have gained attention. The IAEA has established basic principles for exemption that limit the radiation dose that individuals or population groups may receive from exempted practices or sources.

This paper discusses the recent IAEA Advisory Group's recommendations on principles for radiation practices and sources in the recycling of retired components and materials from nuclear facilities. The background of the Advisory Group's work is discussed, then its methods and preliminary recommendations are summarized. Finally, a similar effort sponsored by the Commission of the European Communities is summarized and compared to the IAEA approach.

DEVELOPMENT OF IAEA ADVISORY GROUP RECOMMENDATIONS

As the result of the recent IAEA Advisory Group efforts in exemption, in 1987 the IAEA published interim general principles for exemption of sources and practices that may result in both individual and collective doses of very low significance (IAEA 1987a). The principles were quite general and applicable to any type of manmade radiation source that may give rise to trivial risks. They did not, however, apply to natural sources of radiation. The Advisory Group concluded that a trivial risk from radiation exposures would be in the range of 10 to 100 μSv (1 to 10 mrem) per year. Further, the IAEA interim exemption principles were intended to provide a safety margin for selected individuals who may be exposed to radiation from several exempted sources and to account for the uncertainty of future human activities. With this safety margin in mind, the IAEA recommended that the individual doses from a single exempted source or practice should not exceed 1% of the existing individual dose limit for members of the public, or 10 μSv (1 mrem). This dose equivalent is less than 0.5% of the annual effective dose equivalent from natural background radiation and is small compared with the natural variation in background radiation. For skin doses, the IAEA Advisory Group recommended a dose limit of 1% of the existing limit, or 500 μSv (50 mrem) (IAEA 1987a).

The 1987 IAEA Advisory Group statement also recommended the control of collective dose. This would provide additional assurance that many small doses would not total a significant dose. It would also guard against the possibility that exemption could occur without the knowledge of controlling authorities. This is especially important for sources that are exempt and, therefore, not subject to notification and registration. The Advisory Group recommended that, as part of the "basic case" for exemption, the collective effective dose equivalent commitment from the exempted source or practice be on the order of 1 manSv (100 manrem) or less (IAEA 1987a). The 1987 interim guidance attempted to account for all potential dose arising from an exempted source or practice over all time. National authorities could exempt sources that give rise to larger collective dose commitments, but could also establish a condition below which no further consideration need be given to the radiological basis for exempting a source. The Advisory Group considered that sources and practices complying with the conditions relating to individual and collective dose may be exempted from the normal regulatory requirements of registration and notification, and treated as if no radiation exposures were involved.

The IAEA received numerous comments on the interim exemption principles from the participating countries. Many of the comments related to the proposed collective dose criterion, which normally was used only as a tool in the optimization process. Applying this criterion was considered to be overly restrictive and was limiting for many of the cases under review. Another major issue was the definition of a practice. This definition is directly related to the collective dose criterion, which creates a problem, because the numerical criterion takes on a very different significance depending on how a practice is defined. For example, if the practice of exemption for recycle means application to the material from a single reactor, the derived limits would likely be quite different than those derived where the practice is defined to cover the material potentially recycled from all reactors in a country.

Because of these concerns, it became evident to the IAEA that further discussions were needed if a firm international consensus on exemption principles were to result. In response, the IAEA called an Advisory Group meeting in Vienna, Austria, in March, 1988, to modify the interim statement. The Advisory Group was co-sponsored by the IAEA and the Nuclear Energy Agency (NEA) of the Organization for Economic Co-Operation and Development (OECD). The Advisory Group prepared a guidance document that recommends a policy on exemption (Linsley and Salo 1988). The IAEA policy was later published as Safety Series Report No. 89 (IAEA 1988). This policy calls for the system of notification, registration, and licensing prescribed in the IAEA Basic Safety Standards for Radiation Protection (IAEA 1982). Safety Series No. 89 allows exemption on a generic or case-specific basis. The revised definition of a practice for exemption purposes is "a set of coordinated and continuing activities involving radiation exposure which are aimed at a given purpose, or the combination of a number of similar such sets" (IAEA 1988).

The Advisory Group further clarified the basis for exemption by identifying two basic criteria to determine a candidate for exemption: 1) radiation protection must be optimized (as shown through a cost/benefit analysis) and 2) individual risks must be sufficiently low. In determining a trivial level of individual dose, a trivial risk level must be chosen, and average natural background levels should be considered as a reference level. The Advisory Group concluded that most authors identify trivial levels of risk to be in the range of 10^{-6} to 10^{-7} per year (IAEA 1988). Using a risk factor of 10^{-2} per Sv (10^{-4} per rem), a trivial dose range of 10 to 100 μSv (1 to 10 mrem) per year results. Comparison with natural background gave a trivial effective dose equivalent range of 20 to 100 μSv (2 to 10 mrem) per year. The Advisory Group concluded that, regardless of the source, a trivial level of dose could be assured if it did not exceed the order of some 10s of μSv per year (IAEA 1988). Accounting for the potential for exposure to many individual exempted sources, critical group doses on the order of 10 μSv (1 mrem) per year would be reasonable.

The Advisory Group further concluded that optimization of protection must be considered analogously to the requirements in the IAEA basic safety standard (IAEA 1988). The optimal level of protection is achieved when the next spending level exceeds the health detriment averted. Using the IAEA minimum value of $3,000 per manSv in 1983 dollars (IAEA 1985), a source-earelated trivial collective dose for exemption on the order of a few manSv results. For continuing practices, the Advisory Group interpreted that a commitment of about 1 manSv (100 manrem) per year of practice would be reasonable. This revised approach accounts for the 50-year dose commitment per year of practice instead of the dose

commitment over all time. The Advisory Group's acceptance of the concept of optimization allows a previously prohibited degree of flexibility if individual risks are appropriately low.

DESCRIPTION OF GENERIC IAEA METHODS

The main objectives of additional IAEA Advisory Group efforts to develop generic exempt quantities have been:

- to illustrate a methodology for developing practical radiological criteria through the application of the IAEA preliminary exemption principles,
- to establish generic criteria, and
- to determine the practicability of the preliminary exemption principles (IAEA 1987b).

Exempt quantities, expressed in units that relate to radiation-detecting instruments, are a more practical expression of the general exemption principles. The steps used by the IAEA Advisory Groups in deriving exempt quantities for a defined source or practice are:

- to establish a series of radiation exposure scenarios that account for various exposure pathways and conditions,
- to estimate the resulting radiation doses to individuals and population groups for these scenarios,
- to determine the limiting (highest dose) scenario for each radionuclide, and
- to determine the concentration of individual radionuclides that would result in the exemption criteria (dose limits).

In assessing the radiation doses, the IAEA Advisory Group advised careful selection of parameters, assumptions, and data (IAEA 1987b). For their assessments, the IAEA Advisory Group and their consultants judged scenarios on the likelihood of their occurrence leading to human exposure and on the likely magnitude of those exposures. In addition, estimating the potential exposure of a critical population group was necessary.

In their evaluations to date, the IAEA Advisory Groups considered:

- disposal of exempt wastes in a sanitary landfill (IAEA 1987b);
- disposal of exempt wastes by incineration (IAEA 1987b);

- recycle of contaminated steel, aluminum, or concrete**; and
- reuse of concrete buildings, tools, or equipment.**

Because of the potential value of scrap materials recovered during decommissioning and because of national and international efforts to minimize wastes, exemption criteria for recycle and reuse have gained recent attention.

The radiation exposure pathways included in the generic dose estimates were external exposure to penetrating radiation, inhalation of airborne material, and ingestion of contaminated foods or removable surface contamination (through secondary transfer from hands to the mouth). A variety of representative radionuclides were considered to explore fully the radionuclide-dependence of the resulting exemption limits. These radionuclides were chosen to represent alpha emitters (^{239}Pu and ^{241}Am), high-energy photon emitters (^{60}Co), low-energy photon emitters (^{55}Fe), and pure beta emitters (^{90}Sr and ^{99}Tc).

The potential radiation exposures resulting from different generic scenarios have a probability of occurrence that may range from zero to one. Thus, the IAEA used their collective judgment to select scenarios for the derivation of exempt quantities. They paid most attention to those scenarios where individuals could have direct contact with the radioactive materials. These scenarios included workers at landfills, incinerators, smelters, or recycle centers, and consumers who may use recycled materials or who reuse released buildings, tools, or equipment. Additional scenarios, such as use of ground water near a landfill or release of volatilized material through the stack at a smelter, were also included to provide an estimate of the likely collective dose.

IAEA RESULTS AND DISCUSSION

Example results for the groupings of reference radionuclides and the various types of exemption considered by the IAEA Advisory Group are summarized in Table 1. The results are presented in terms of reasonably expected ranges, based on the various radionuclides in each group and on the expected variation among exposure scenarios. Control is lost in the fate of exempted materials through unrestricted release. Thus, material exempted for recycle or reuse could be disposed of in a landfill, or material exempted to a landfill could be recycled or reused. Because of the lack of future control, the proposal has been made that a single exempt quantity should be established that would cover all alternative future conditions, without specifying limitations for landfill disposal, incineration, recycle, or reuse (Kennedy et al. 1988). This overall limit appears to be possible because of a close grouping of the results shown in Table 1 across most radionuclide and unrestricted release categories.

** As described in a draft working document on "The Application of Exemption Principles to Wastes from Decommissioning and Recycle of Materials from Nuclear Facilities," by the International Atomic Energy Agency, Vienna, Austria, 1988.

TABLE 1. Example Exempt Quantities for Various Exemption Categories

Exemption Category	Alpha	High-Energy Photon Emitters	Low-Energy Photon Emitters	Pure Beta Emitters Emitters
Sanitary Landfill (Bq/g)	1 - 10	0.1 - 10	(a)	300 - 600
Incineration (Bq/g)	0.1 - 100	0.5 - 10	(a)	10^2 - 10^3
Recycle(b) (Bq/g)	1 - 10	1 - 10	10^4 - 10^5	40 - 300
Building Reuse (Bq/cm)	1 - 5	0.004 - 1	10 - 100	60 - 500
Reuse of Tools and Equipment (Bq/cm)	10 - 100	10 - 100	60 - 500	10^2 - 10^3

(a) No radionuclides were considered for the scenarios shown.
(b) For recycling of steel, aluminum, or concrete rubble.

Three potential groupings of released material are: 1) mass concentrations (in units of Bq/g), 2) surface contamination in buildings (in units of Bq/cm), and 3) surface contamination on reused tools and equipment (in units of Bq/cm). Further, it appears that a set of radionuclide groupings could be made by combining the high-energy photon emitters and alpha emitters into a single grouping across all categories. The overall exemption limits resulting from such re-grouping are summarized in Table 2 (Kennedy et al. 1988).

Again, ranges of values are shown to denote the potential variations of radionuclides and exposure conditions. For mixtures, the sum-of-fractions rule could be applied. The net result for mixtures is that the limit is controlled by the most restrictive radionuclides present. It may be noted that for all scenarios considered, the individual dose criterion is limiting in relation to the exempt mass concentration and surface contamination limits. However, the collective dose criterion is the determining factor in assessing the total quantity of material that may be buried, incinerated, or recycled. The IAEA work will continue because further studies are needed to determine whether additional practical considerations (including costs and detectability) will change results or basic conclusions.

TABLE 2. Preliminary Exemption Limits for all Release Categories

Limit Category	High-Energy Photon and Alpha Emitters	Low-Energy Photon and Pure Beta Emitters
Mass Concentration (Bq/g)	1 - 10	10^2 - 10^3
Building Surfaces (Bq/cm)	0.1 - 1.0	10^2 - 10^3
Reuse of Tools and Equipment (Bq/cm)	10^1 - 10^2	10^2 - 10^3

OTHER INTERNATIONAL EFFORTS

In parallel with the IAEA efforts over the past few years, European countries have adopted national policies regarding exemption. In 1988, the Commission of the European Communities (CEC) published Radiation Protection No. 43: *Radiological Protection Criteria for the Recycling of Materials from the Dismantling of Nuclear Installations* (CEC 1988). This document is a summary of the recommendations from a group of experts set up under the terms of Article 31 of the Euratom treaty. The CEC identified two approaches to the establishment of radiological protection criteria for the recycling of materials from the dismantling of nuclear installations. One is based on setting acceptable individual and collective dose levels (similar to the approach adopted by the IAEA), and the other is based on setting clearance levels for the activity concentration of the materials considered so that the radiological consequences of recycling are insignificant from a health protection point of view. The CEC concluded that the second approach, setting clearance levels, had the advantages of regulatory simplicity, practicability, and consistency and afforded the same level of health protection as the first approach (CEC 1988). For more complicated cases, it would still be possible for national authorities to carry out a case-by-case assessment. Because of these considerations, the CEC adopted the clearance-level approach; however, they also provided that the IAEA dose limits would be met for specific cases where alternative criteria would need to be established. In addition to the IAEA individual and collective dose limits, the CEC also considered the IAEA transportation safety regulations limiting the surface activity for materials.

The group of experts to the CEC considered the results of a study on radiological protection criteria for recycling carried out by an ad hoc working party. This working party considered radiation exposure scenarios for steel scrap and equipment from dismantling retired nuclear power reactors. The recommended clearance levels for recycling steel scrap and equipment are summarized in Table 3. It should be noted that the 0.4-Bq/cm^2 limit for non-fixed beta-gamma emitters is derived from the IAEA transportation safety regulations.

TABLE 3. Clearance Levels Recommended by the CEC for Recycling Steel Scrap and Equipment

Type of Radiation	Mass Activity Clearance Level	Surface Level Clearance Level
Beta-Gamma	1 Bq/g averaged over a maximum mass of 1000 kg[(a)]	0.4 Bq/cm^2 for non-fixed contamination on accessible surfaces[(b)]
Alpha N/R[(c)]		0.4 Bq/cm^2[(b)]

(a) No single item may exceed 10 Bq/g.
(b) Averaged over any area of 300 cm^2 of any part of the surface.
(c) N/R means that no value is recommended because of potential detectability problems and because most alpha activity is assumed to be present as surface activity only.

SUMMARY

The IAEA's individual dose criterion is selected based on a consensus of what constitutes a trivial risk and a comparison with natural background. The collective dose criterion established by the IAEA is intended to serve as the basis for an optimization assessment that can help justify exemption decisions. The IAEA is in the process of developing exempt quantities for recycle and reuse using a radiation exposure scenario analysis. The exempt quantities will be useful to illustrate a methodology for developing practical radiological criteria, establish generic criteria, and help determine the practicability of the preliminary exemption principles.

As an extension of the IAEA exemption principles, the CEC has derived clearance levels that define the activity concentration of the materials to be recycled, based on a generic radiation exposure scenario analysis. Both the IAEA and the CEC efforts indicate that national authorities have the flexibility to modify the overall approach to account for special situations that may require additional regulatory actions.

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Summary & Panel Discussion

*Summary & Panel Discussion**

Moderator: Anthony B. Wolbarst (EPA)
Session chairmen: Donald A. Cool (NRC)
Masao Oshino (JAERI)
Allan C.B. Richardson (EPA)
Andrew Wallo, III (DOE)

DR. WOLBARST: The panel will be soliciting comments and questions from the audience. But before that, we are scheduling something extra which will take a few minutes, but which I think will be of general interest.

In his opening address, Rich Guimond stressed the importance of effective risk communication in dealing with the public. Risk communication is an issue that is taken seriously by us at EPA. We have established a Risk Communication Program, for example, involving people with expertise in the subject, to provide assistance and guidance for programs with public relations problems. And we have set up a two-day course on risk communication that 10 percent of all EPA staff eventually will take.

An important contribution to this effort is a book, "Improving Risk Communication," recently published by the National Academy of Sciences (NAS). This document was funded partly by EPA and several other Federal agencies. The primary staffer at NAS who handled it was Rob Coppock. So I am going to ask Rob to take a few minutes to describe the book, and perhaps tell us how we can get it.

DR. COPPOCK: Rich Guimond pointed out that in the rule-making process it is important not only to provide the technical means to ensure protection, but also to instill public confidence in it. And that is very much what this report, "Improving Risk Communication," is about.

Yesterday I saw a memorandum signed by Secretary Watkins of the Department of Energy that is going to the heads of operating units, divisions, and programs in DOE, accompanying copies of this report. Secretary Watkins stated that he intends to make improved risk communication a Department priority, and asks for their response as to the changes in

* Much of the tape recording was barely audible, and we had to do a fair amount of heavy editing. Hopefully there are no serious mis-representations. The Editors.

procedures they would expect to implement as a result of the contents of the report. That is a pretty powerful statement and endorsement from his Office.

Also, Terry Davies, the Assistant Administrator at EPA for Policy and Evaluation, is distributing about 250 copies of the book within EPA. The Toxic Substance Disease Registry, the Public Health Service organization responsible for preparing the hazard profiles at Superfund sites, purchased enough copies of the book for all its field offices.

So this is a document, I think, that you will be seeing around from time to time. The reviewers have stated that they expect it to be the standard reference in the area.

I want to take 2 minutes to summarize this 200 page document, and to point out a couple of the conclusions that are relevant to the issues we have been discussing here.

First of all, the book emphasizes that there is no single risk communication problem, so there is no simple solution. There is no silver bullet. Further, improvement will take much effort over many years, not months. We are in it for the long haul.

Secondly, improving risk communication will not necessarily smooth management, making things easier. In fact, it may make things more controversial. Many of the issues of concern are ones on which people have different opinions, use different vocabularies, hold different values regarding the possible outcomes. That being the case, clarifying the issues may make people realize that they have very strong positions, and are in conflict. Helping people understand the issues may increase the level of conflict. The report states that improving risk communication may make management issues more difficult. But we are obliged to make things clearer, even if it makes our jobs more difficult.

Lastly, the report points out that both technical competence and communication skills are needed in communication programs. You will shoot yourself in the foot if you get either one wrong -- there needs to be a balance.

Good communication skills need to be applied early in the life of a project. Very often the communication and presentation problems can only be solved by adjusting analysis and management steps along the way. Unless you have included what people are worried about in your analysis, you may present your conclusions, and find them rejected because they don't include the things that people are really worried about. If people do not see the issues that they believe to be important in the analysis, they are likely to dismiss the whole analysis.

The book is available, and can be ordered from:

National Academy Press
2101 Constitution Avenue, NW
Washington, D.C. 20418

paperbound $29.95
hardbound $39.95

(5-24 copies 15% discount, 25-499 copies 25% discount)

DR. WOLBARST: Thank you, Rob. We are running a bit late, so I feel we should start the Panel discussion. Let's begin with Andy Wallo, of DOE, who chaired the session on "Impacts of Cleanup Technologies and Economics on Criteria".

MR. WALLO: We have been asked first to summarize the session that we chaired, and so I will mention the papers that were covered in the session on impacts of clean-up technologies and economics on criteria. There were seven informative papers.

The first three papers covered limitations of clean-up technologies. They talked about procedures and methods of decontamination, particularly of portions of facilities, buildings, and surfaces, and for measuring the levels of residual radioactivity.

A fourth paper discussed the processing of low-level radioactive waste, and criteria for release. This one looked mainly at reducing volume, again on wastes from buildings and decontamination projects going for disposal.

The fifth paper discussed the disposal capacity and projected waste volumes within the compacts, and another described the Superfund volume reduction process for specific sites in New Jersey.

The final paper, on residual radioactivity cost impacts evaluation looked at various options for controlling contamination and for decontamination standards, again of structures, and at the effects they might have on the cost of cleaning up the structures. The costs of these will be impacted by the criteria, the options selected for decontamination -- whether it be decommissioning a building or totally destroying and removing it. Cost could be impacted by the availability of a BRC limit and the amounts of the wastes.

I think an important message in Bob Dyer's paper on volume reduction was that, for soil criteria clean-up, we have two options: either remove the contaminated soil or leave it in place. Volume reduction was looked at on a very site-specific basis. It was for one radionuclide and one site, and would have to re-evaluated again every time one wanted to do it. It is not really a generic process for volume reduction; its applicability depends on the criteria that apply for cleanup, and the dose levels.

It is clear that both the actual clean-up and the measurement processes are going to be impacted by the criteria selected, and that a site must be characterized before the remedial action. When you are deciding on your final cleanup standard, you really want to have a complete characterization of the sites.

It is essential, at the end of a project, to document the clean-up and the procedure used in verifying or certifying that the site meets standards. Measurement techniques and detectable limits vary, and the standards and associated certification procedures should be tied to the risks and benefits of the project. In almost all instances where statistical sampling procedures are used for certification, another survey could measure something that is not measured with the procedure used. While these differences in results are not likely to have any significance in the

overall risk associated with the site, they could cause questions in the validity of the sources. A certification process has to be tied to a procedure for measurement.

And again, the issue of whether BRC standards affect clean-up, and whether volumes can be averaged over mass, or surface contamination averaged over mass, can impact decisions and the costs of clean-up.

Some of the discussion that we might want to follow up on concerned the general tendency in a lot of States or communities to say that anything that ever was touched with radioactivity is radioactive waste. We have seen that in a number of our DOE sites. Some barrels, dug up near one of our sites, were filled with sludge and petroleum-based residues that were actually associated with an airport construction action. An analysis was done on the wastes, and some radium and uranium showed up. It was clearly part of the petroleum-based material, and was not radioactive waste. But the analysis showed radioactivity, and the headlines said that this was radioactive waste that approaches or exceeds NRC unrestricted use standards. So here we have a solvent or a petroleum-based material that nobody is going to drink. But we are looking at drinking water standards or unrestricted release standards and calling it radioactive waste, because it has this taint of radioactivity in it. Clearly, the hazard associated with this material was chemical.

DR. COOL: The last session that we had yesterday afternoon was entitled "Health Effects," although it ran the gamut on a number of different issues.

The first paper by Bill Ellett talked about health effects predictions and analysis. It provided us with a clear view of the very unclear view we have of radiation risks. Although we have a lot of information, there are great uncertainties. And the doses that are involved with residual radioactivity assessments, and the levels that we are actually finding, are well below the levels where effects have been demonstrated. In fact, the risks are likely not to be measurable in the population with the statistical power we have in the groups available.

Mr. Yamamoto provided a paper on JAERI's experience in decontamination and reuse of a large-scale radiochemical laboratory, indicating some of the criteria that were actually applied in a live situation and some of the doses actually resulting. One of the things that impressed me was how low the doses were when they were finished with the decontamination, on the order than less than a tenth of a micro-Sv. That corresponds to an extremely low level of potential health effect and risk.

We then had a couple of papers dealing with exposure models, the first by Don Lee on applied exposure modeling for residual radioactivity, and the second by Andy Wallo on the DOE guidelines.

Modeling is an area that we end up using whether we like it or not, because we cannot empirically know all of the things that are going to happen prior to actually releasing the site. The point was made that modeling can be supportive of rulemakings, compliance determinations, and research, and that those three activities have very different goals and require very different types of models.

Site-specific models can provide an accurate means for dealing with case-by-case determinations, but they have a potential drawback of discrepancies between sites, and between different actions at different times. The alternative to that is generic models. These are useful in rulemakings by a regulatory agency, like the NRC, doing a regulation which would involve a release for unrestricted use. But models of that type are not very realistic when you start looking at the details of an individual site. A model which is generic is not going to represent the western desert of the United States very accurately, just as it is not going to represent the eastern United States very well. The appropriate level of detail may be difficult to achieve with a generic model. And one of the underlying themes in this conference is that for models to be useful, they need good input data. And that requires an accurate characterization of sites before anything else is done.

MR. RICHARDSON: The next session was a monstrous one, with 12 papers. I hope I don't neglect any of the major points of the various authors, because we have so many to talk about.

Rob Coppock started it off, pointing out that there are four kinds of national regulatory styles. He called them evidential, consensual, authoritative, and corporatist. What I found most interesting is that even though all these four styles may exist, all of the countries end up with basically the same kinds of answers to radiation problems. They all use the same basic decision criteria. So the conclusion for me is that it does not matter how you get to the decision. If you use the right principles, you will probably end up with pretty close to the right decision.

After Rob's paper, I reviewed the basis for such decisions. They are the old radiation protection principles we have had with us for many years. There are two new factors that are important, however. One is national (in this country) mandates under legislation for Superfund for ground water protection levels. The other is more conceptual, the need for source-related categorization of the individual dose limit. You cannot use the whole 100 millirems for residual radioactivity. Then I raised the issue of whether we should be looking for dose limits or for concentration and quantity limits for our actual criteria.

Vern Rogers carried that much further in his paper, and suggested categories of nuclides -- a number of other authors did the same thing. He also emphasized the importance of individual dose scenarios and their large uncertainties.

Mr. Yamamoto and Mr. Oshino reviewed the Japanese situation. They are using some of the IAEA recommendations for exemptions, and are at least headed in the direction of using those same criteria for residual radioactivity standards. (The IAEA exemption criteria are on the order of a millirem (0.01 mSv) for individual dose, and 100 persons - rems (one man-Sievert) for collective dose.) They are not using the collective dose criterion.

Lynn Wallis described the decommissioning of Shippingport. I think the most important thing that came out of that experience is the finding that they would pass most of the criteria that have been discussed as site release criteria, and are well below the levels of residual radioactivity that they might have to meet. None of these levels have been set yet, of course, but Shippingport would have passed almost any of the proposals. So it can be done. One of the

interesting sidelines of the paper was the discussion of how ALARA was implemented at that site, and the importance of social factors.

Stan Neuder discussed methods for deriving surface contamination limits, Reg Guide 1.86, and associated problems. A number on papers touched on the difficulties with Reg Guide 1.86. Clearly, there is lots of work to be done on surface contamination criteria.

Tim Johnson and Don Cool described NRC activities. The NRC is moving ahead aggressively with initiatives based on their legislative mandates for low-level waste, and also on initiatives for "below regulatory concern" or "exemption levels", depending on what lexicon you use. Their exemption policy, however, is at variance with that of the international community. It is an order of magnitude higher -- 10 millirems per year, as currently proposed. Similarly, the criterion that was proposed for exemptible collective dose -- a bottom line, as it were, for when you have to look at ALARA in detail -- is a factor of five higher than the IAEA recommendation. So that is something that, I am sure, will be a discussion item in the future.

Steve Adams talked about the tortuous history of ANSI N13.12. I was glad that we had his presentation, because perhaps the regulated community, after having gone through that exercise, has gained some perspective on why regulatory agencies take so long to get the answers out. And when they are finally out, why they are generally perceived by the public and the industry as arbitrary and capricious. [Laughter]

Bill Holcomb described EPA's draft proposed low-level waste standards. These standards have two significant new things in them. One is that for the first time they reflect national ground water protection policies, which the current standards do not. Also they reflect EPA's (at least up to now, EPA's) approach to BRC matters, and that is to do it on a case-by-case basis and feel our way along on the question of BRC.

The last two papers, by Jim Vance and Joe Ray, dealt with BRC. Jim distinguished three related concepts: de minimis, BRC, and generic BRC. He pointed out the difficulties with generic BRC. They are not unrelated to the kinds of difficulties that caused EPA to be cautious in approaching BRC, and described their petition on BRC for specific waste treatments.

I was interested in the extremely low population impact of the proposed exemptible waste streams. Three hundred man/rems is only about a 10 percent chance of one cancer in 10,000 years.

Joe Ray also talked about the problems of Reg. Guide 1.86 for uses down the line, and possible alternate uses for decommissioned facilities. He pointed out the need for basic dose limits and accepted pathway models.

To summarize the whole session, at least for this listener: We don't have the answers with respect to residual radioactivity criteria, and we need them. There seems to be general agreement on the need for a dose limit, if not as a single standard, at least as a guidepost that guides all the other standards.

This dose limit could possibly be BRC, and we need to explore carefully the possibilities for implementing such a limit in a sensible way. Alternatives that have been suggested are

standardized models, pathway models, or standardized results based on conservative scenarios expressed in terms of concentration levels.

MR. OSHINO: The last session covered the recycling of materials and equipment. In this session, four papers were presented. The first, by Mr. Nakamura, concerned a research program on the recycling of decommissioned materials at JAERI. The decommissioning of JPDR started to demonstrate real reuse of materials. Such a program is now in progress.

The second paper was on the effects of residual radioactivity in recycled industrial materials. Mr. Kato presented this paper, and concentrated on the effects of residual radioactivity on LSI and photographic films.

The third, by Mr. Kennedy, was on the development of international exemption principles for recycle and reuse on the sites. He presented IAEA's exemption principles.

The last one, by Dr. Lichtman, discussed DOE's practice of using the National Environmental Policy Act, until cleanup criteria are established, to decide recycling issues.

DR. WOLBARST: Thank you. I want to thank the chairmen for their summaries, and open the proceedings for discussion, comments, questions.

QUESTION: Would you say that EPA people here today have come away from this workshop with a consensus or some conclusions on the next steps, and if so, what are your reflections on this?

DR. WOLBARST: I think it is not a matter only of what EPA people think, and I would like to ask all the members on the panel to respond.

MR. RICHARDSON: We need a dose number, and we need a way to implement it.

We also need to explore the question of generally acceptable models. That is going to be a key to getting numerical criteria. There are a number of models out there now, and one of the things that is on my list is to see if we cannot get some agreement on some common agency models from EPA, NRC, and DOE.

And we should explore the idea of BRC as a basis for numbers. It is not clear to me that BRC is the answer. One of the problems with it is that there are going to be some obvious situations where BRC limits cannot be reached, not the kinds of BRC limits that the general public can accept. So we have to look at those situations and see how we are going to handle them.

One alternative is a relatively high individual dose limit, high compared to the BRC, at least, and the application of ALARA. That is the current DOE approach, as I understand it, but that needs to be tested, too.

DR. COOL: The general approach which is being pursued is to try to establish some dose numbers (although we do not necessarily know what the numbers are going to be) and from there move to generic criteria that would be useful in screening situations and site-specific criteria, to allow for individual licensees or users who wish to be more sophisticated. This approach seems to have a great deal of support around this room.

This feeling may be coming more from the discussions we have had over breakfast, lunch, and dinner than from the papers, but I have heard a great deal of support for the general approach which the NRC is taking, which Japan is taking, and others, in setting up a structure which starts from the basic dose and risk situation and then moves to values which can be implemented, measured, and complied with on a site-specific basis.

So from my perspective (and not necessarily that of the NRC), I come away from this conference with some encouragement that we are moving at least in the same general direction.

MR. WALLO: I was glad to hear what Allan Richardson said about BRC. I did have a bit of concern when we started talking about BRC. Bill Holcomb talked about it with respect to low-level waste, and we then automatically looked at applying a low-level waste BRC to decontamination and decommissioning of structures.

I think we must be careful when we consider BRC or some other number plus ALARA. BRC is basically, or should be, I think, a case-specific or source-specific determination. A generic BRC for everything does not seem to me to be a thing that we can sell very readily. BRC for low-level waste should be different than BRC for cleanup. Ratios of individual risks and population risks (collective dose) are likely to differ. We need a BRC approach to balance these.

When you are doing decontamination and decommissioning, you are left with residual radioactivity. You have soil and buildings and equipment that you have to clean up, and those may be disposed of in a landfill, or released for reuse or to be salvaged, and you have no control over them once they are gone.

These are all very different scenarios (exposures associated with the site being cleaned and exposures associated with the waste disposal), and they may not necessarily have the same BRC risk associated with them. It may not be possible to get soil criteria for remedial action down to the same level that you might get releasable, recyclable material.

So I think we want to be careful, at least in the beginning, in considering BRC. As an alternative, we could look at an upper limit, as Allan suggested, and then try to do site-specific analyses. That is another option. I think it is good to step back and look at all of those before we decide on where we are going.

The screening criteria approach is always interesting, and it is desirable to have a screening level so that you can say, 'If I am below this, I am okay, and if I am not, then I have to do a site-specific evaluation, so that I can clean up to meet a limit or get below it.' The drawback we are faced with is that those screening numbers many times get written in concrete. Can we afford to set screening levels effectively and still offer the alternatives of doing site-specific evaluations where it is necessary?

COMMENT: In preparing the closure plan for our low-level waste sites, a technique like what you just described was used. We used the environmental impact statement and 10 CFR 61, where it gave a generic scenario for both a humid and a dry site, and then defined within those scenarios what would be allowable limits for contamination on the buildings, equipment, and the soil. If your contamination levels are at those concentrations or below, you don't have to go through any further site analysis; you can go ahead and release your site for closure. However, if you have greater than those concentrations, then you do further analysis. So there is one instance where there are dose limits and derived concentrations, and with just a few criteria you could compare your site with a standard.

MR. WALLIS: I might mention a problem we have run into several times. We closed the cooling towers down at a site in California, and because there wasn't any BRC number we could use, we simply had to declare everything low-level waste, although we knew it was not low-level waste. The cooling tower material was eventually sent to conventional burial ground; it was not declared LLW. At Shippingport, we had a lot of asbestos, and we knew it was not contaminated, but we had no choice but to declare it low-level waste.

So there is a need for BRC in those kinds of situations. And without it, you cannot take any chance at all. You just simply declare it low-level waste. We are talking about large volumes.

DR. LEE: I have a whole list of things to comment on. I heard the call for a generally accepted model. As a modeler, I am very sensitive to the fact that the scenarios and the approaches must be agreed upon, but I think that, as part of a competent risk management-risk communication process, the burden of proof is always on the person attempting to do the study or to run the model. That requires adequate documentation of all steps; adequate QA of the model and its development, adequate QA of each of the scenarios and adequate QA of the simulations that are run. This is necessary to build the credentials of the results in a manner necessary to have them believable both to the technical community and to the public.

I think that selecting a single model in some regards may be a way to streamline that, but it may also have the pitfall of trapping everybody into having a single model that may be inadequate for specific situations, and I would caution that I am not an advocate of a single-model idea.

DR WOLBARST: Do you support the idea of the Federal model that we raised a couple of years ago? We held a meeting of a number of agencies at EPA to discuss the possibility.

DR. LEE: Modelers have a variety of different attitudes. I view most models as being a collection, sort of a bunch of little models put together to form one giant black box. I am not an advocate of that approach because I think that the individual components need to be tailored to match the physical or chemical situation encountered. If you have one black box, it is very difficult to prepare a site-specific analysis that can accomplish that.

I think the message that I like, that I heard over dinner last night, revolves around the risk communication points. Maybe it would be simpler to try to communicate a single exemption number that would cover BRC and all types of releases. That is not to ignore the fact that might justify separate practices at different levels of risk because of the associated cost. My point of view, rather, is to look at the assignment of different levels of risk for separate practices very hard before you attempt to do it. In saying that 4 millirem is okay for drinking water and for BRC waste, and 10 millirem is okay for recycle/reuse, it may be very difficult to communicate the basis and the health risk impacts of each of those kinds of practices, sources or activities. It may be better to have a single level that is communicated across the board in a uniform manner. Allow exemptions for whatever X millirem equals for waste stream A or waste stream B or recycled material components D or E, and let the analysis determine the screening levels or the derived criteria based on a single dose standard.

MR. RICHARDSON: One of the pitfalls that I think we need to be careful to avoid is to set the BRC level so high that the public does not accept it as BRC. That is a real danger. If you do the arithmetic and calculate the risk factor for a number like 20 millirem, you come out with risks that are on the order of 10^{-3}, and nobody is going to accept that as a tolerable risk.

So if we are going to have something which will be accepted by the public as being a trivial risk, so that we can throw the stuff away and walk away from these little tiny bits of contamination, it has got to be truly trivial.

On the other hand, if we choose numbers that are truly trivial, there are going to be cases that will not fit, that it will not make sense to clean up to that level. I think the problem that is before us is to figure out how to deal with those situations in a rational way.

One of the alternatives that we have not talked about at all is various forms of institutional controls. Another possibility is an exception clause where we have to dig too deeply into the public pocketbook: 'It is not worthwhile doing this, we cannot afford to do it.'

There are a variety of ways to handle these problems, but I think we have to recognize that there are two classes of problems: One class of problems is getting public acceptance for below regulatory control, and the other class of problems is dealing with the hard cases.

COMMENT: I want to pick up on that, if I may. De minimis is one thing; below regulatory concern is another, and there are still some controls associated with below regulatory concern as viewed in some way. It is not an uncontrolled situation. The phrase is a terrible phrase, as we all have said. I don't know how to communicate to the public when we are using a phrase

that they will not be able to understand. Perhaps we need to do some work on that in its own right to help communicate what we are really trying to achieve with a phrase of that nature.

From the electric utility perspective, we are concerned about the large volumes of materials that would normally go to a disposal ground that do not have high levels. We are not interested, necessarily, in the recycled part. Our interest is more the volume aspect, and being able not just to eliminate any control, but to control it in a way that is acceptable.

Andy Wallo expressed some concern about a situation in which you take material to a sanitary landfill, and have to go back and clean up the sanitary landfill at some later time if the ground rules change. We have certainly been exposed to that under the Superfund area. That would suggest that maybe we don't go to sanitary landfills, but maybe we do have to have other options.

COMMENT: If you think about the whole subject and these last 2 days of discussion and of trying to set either exemption or BRC levels, it runs counter to what people have viewed as prudent health physics judgment, if you will, associated with control of potentially radioactive materials.

If you cannot measure the contamination on the surface of a component that you want to release, or you have such a large bulk of material that it is very difficult to certify it as not being contaminated, the prudent health physics practice has been to declare it low-level waste, independent of the volume or the cost or whatever, and send it to disposal. So what we are attempting to do now is to reverse what in the past we have called this prudent health physics judgment, so as to allow a different regulation to come into effect, to reduce the volumes of material and make decisions that may in fact run counter to historical practice. Of course, there are going to be problems in implementing it, and of course there are going to be these questions about 'Well, gee, does this mean you have to go and clean up sites that you are going to have a tough time identifying from the records that are pretty much non-existent? Will we always be stumbling over new sites to be cleaned up?' And the answers are likely to be 'Yes'. We are going to have to commit to some additional remedial action kind of program if this is going to fly.

DR COPPOCK: As the most extreme "outsider" in the room, not being from the health physics profession at all, I am rather astonished that you think it possible to dismiss sites that had been dealt with in some way in the past. With regard to hazardous chemicals, we have embarked on revisiting old decisions. Just because it was the best we could do 20 years ago does not mean it is the best we can do today. We ought to treat old hazards with no lesser standards of safety than we treat new hazards today.

COMMENT: With respect to depending on Superfund to solve the future problems that we are seeing, faced with triple damages under Superfund, I am sure that the prudent decision would probably be to work with the NRC to try to resolve it without having to go through that kind of process.

DR. NEUDER: I was going to ask one question, but let me combine it with another. I was going to ask whether particularly the EPA group had thought any more about whether standards or guidance were appropriate for these kinds of situations.

Now I got led back to something that we also thought about a few years ago that was just brought up. Grandfathering is a major issue. If you set stringent requirements now, it is very difficult to keep people from the demand for grandfathering. It is very hard to write words that will convey the notion that it does not pay to go back to many of these sites, that it is close enough; let them alone. I am not talking about ones that are lousy or where things were not addressed that we have subsequently found to be hazardous, but rather where the radioactivity was addressed, and we just cleaned up at one level, and we would all, by consensus, agree that it does not make sense to remount the effort to bring it down to a new level.

So, unless you can solve that problem, that becomes a compelling argument for guidance, because guidance can deal with things like this, very different from standards. In particular, guidance can work the way DOE is. You can set up a 100-millirem limit, and you can say do ALARA and take account of the following kinds of things, because any of these old sites will comply, almost all of them, comply with a 100-millirem limit, and so they don't raise this issue of compelling, of creating an urgent situation on something that doesn't demand it.

COMMENT: I cannot see a situation in which a site was delivering -- let's take the worst case --, in perpetuity, 100 millirems a year to anybody who happened to live on it, not getting on the National Priority List and having to be cleaned up to some arbitrary level, in the absence of more specific criteria than that kind of guidance would provide.

COMMENT: I think the point Stan was trying to make is that there are a lot of these little sites out there that maybe, if we were doing it today, we would have cleaned up a lot neater and left in a lot nicer package, but on the other hand, it is not worth a lot of effort to go back and spend big bucks to even characterize or look at these sites if you have enough documentation in the file to really let them know that we are in the noise-level type of situation.

COMMENT: The federal risk management process is a process by which regulatory decisions and actions are made by balancing a number of factors. Those factors include the health risk potential of the situation, the costs, regulatory compliance, the achievability of technology with keeping people working, and all of these things are weighed and a decision is made in support of a licensing or a regulatory decision.

Now, in the case of an old site, it may be that there is high uncertainty about the actual potential of that site to cause a public health risk. But there may be well-documented factors like the risks of workers digging up old sites that may become a compelling argument in favor of the decision to stabilize it in place, rather than dig it up and move it again. Of the potential alternatives for dealing with that situation, the worst may be to dig everything up, repackage it, and move it someplace else -- and something lesser, like stabilizing it in place, may be in fact an acceptable risk management-based regulatory decision.

COMMENT: If you go back and look at the records at old sites to evaluate whether, because of new and stricter standards, perhaps they need to be cleaned up, you will notice that, invariably, all the records are going to show that the measurements were less than whatever the standard is now.

COMMENT: There are maybe a couple of factors that we have not discussed much, but which impact on all of this. We talked about 4 mrem per year as being different from 10, as being different from 20, as being different from 1. And I am not at all sure that we can support the differences between those numbers when we actually go out and do surveys, simply on the basis of counting statistics, let alone on the basis of the differences in the risks. We talk about people understanding what 4 means. I am not convinced I know what 4 means. Does anybody know what 4 means?

We would all admit that we don't know what the absolute risks are for these dose ranges, which are very small increments over 100 millirems. Whatever the risk of 100 millirems, I think we can agree that the risk of 104 millirems is 4 percent bigger, and the risk of 101 millirems is 1 percent bigger, and so on.